Occupational Risk Consequences of the Department of Energy's Approach to Repository Design, Performance Assessment and Operation in the Yucca Mountain License Application

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REPORT SUMMARY

EPRI has discovered several aspects of the U.S Department of Energy (DOE) proposed design and operation of the Yucca Mountain repository that—if implemented as described in the license application (LA)—could result in unnecessary occupational health and safety risk to workers involved with repository-related activities. This report identifies key DOE conservatisms and focuses on the occupational risk consequences of the DOE's approach to the repository design, performance assessment, and operation.

Background

A deep geologic repository at Yucca Mountain, Nevada, has been proposed for the disposal of commercial spent nuclear fuel (CSNF) from nuclear power plants and other nuclear fuel and high level radioactive waste (HLW) from defense and nuclear weapons programs. The DOE has submitted the LA to the U.S. Nuclear Regulatory Commission (NRC) for approval to construct the Yucca Mountain repository. The LA and its supporting documents present information on the area surrounding the Yucca Mountain site and the design of the proposed repository surface and subsurface facilities. The LA also includes the DOE assumptions and calculations intended to demonstrate compliance with applicable regulatory requirements. Many of these assumptions and calculations are extremely conservative and have the potential to result in activities that could expose workers to unnecessary occupational health hazards. These hazards exceed those that would be experienced if the DOE had developed the design and performed its analyses using a more realistic approach, such as that recommended by the National Academy of Sciences in its *Technical Bases for Yucca Mountain Standards* report issued in 1995.

Objectives

To identify aspects of the DOE-proposed approach to Yucca Mountain repository design, performance assessment, and operation that have the potential to expose workers in the nuclear and other related industries to occupational health risks in excess of those that would be encountered if the DOE had taken a more realistic LA approach.

Approach

In developing this report, EPRI reviewed the Yucca Mountain LA and analyzed 1) the assumptions made by DOE in its analyses, 2) how those assumptions affected the proposed design and operation of the repository, and 3) how the resulting approach has the potential to cause occupational health risks to workers involved with activities at the repository, the reactor, and other commercial sites that could otherwise be avoided if a more realistic approach had been taken. The focus of EPRI's analyses was to identify those activities that could lead to unwarranted occupational health risks and that could be eliminated or modified without impacting the performance of the repository or its compliance with applicable regulations.

Results

EPRI recognizes that there are a certain amount of hazards and risks associated with Yucca Mountain repository-related activities and that it is impossible to reduce such hazards and risks to zero. The term “unnecessary,” as used in this report, is intended to mean the additional risk that may be incurred by performing an activity in the manner proposed by DOE versus the more limited amount of risk that may be incurred by performing the activity in some alternative manner. The difference between the two levels of risk is considered by EPRI to be “unnecessary.”

Unnecessary risks of interest include but are not limited to the 1) proposed use of an undersized transportation, aging and disposal (TAD) canister; 2) exclusion of direct disposal of existing, loaded, dual-purpose canisters (DPCs); 3) underestimation of the fraction of CSNF that will be shipped from reactor sites in a manner that will require processing in a single wet handling facility; 4) overestimation of igneous and seismic hazards, resulting in over-designed facilities and additional complexity for performance assessments and regulatory compliance demonstration; and 5) pileup of conservatisms in assumptions and analyses that have caused DOE to unnecessarily include drip shields in the subsurface design. Any delays in the regulatory process caused by the inclusion of subjects that could otherwise be avoided, or in the shipment of CSNF to the repository, have the potential to impose additional and unnecessary occupational health risks on workers and slowdown in facility completion. Similarly, the performance of any extra manufacturing, transportation, construction, and/or installation activities that could otherwise be avoided carries with it additional health and safety risks for workers. This is especially true for activities involving large and cumbersome components, such as drip shields and transportation casks, or work in difficult environments such as will be encountered at remote sites and in underground locations.

EPRI Perspective

While DOE design and analysis choices, as presented in the Yucca Mountain LA, have led to a demonstration of compliance with the draft Yucca Mountain regulations, EPRI’s analysis has shown that some DOE choices have the potential to cause unnecessary occupational health and safety risks. Such risks could be avoided while still demonstrating repository compliance with the applicable regulations. It is EPRI’s position that DOE should have used more realistic, as opposed to overly conservative, assumptions in designing and assessing the proposed Yucca Mountain repository system.

Keywords

Yucca Mountain
High Level Radioactive Waste
Spent Fuel Disposal

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1
INTRODUCTION

On June 3, 2008, the U.S. Department of Energy (DOE) submitted a license application to the U.S. Nuclear Regulatory Commission (NRC) for authorization to construct a deep geologic repository for disposal of High Level Nuclear Waste (HLW) and commercial Spent Nuclear Fuel (CSNF). The Yucca Mountain license application represents a milestone itself as the culmination of close to two decades of study and evaluation. As a candidate licensee for the construction and eventual operation of a deep geologic HLW repository, DOE has made numerous assumptions and estimates that are conservative in nature. For example, in the January 2008 Total System Performance Assessment – License Application Analysis and Model Report (referred to within this report as the TSPA-LA AMR or TSPA-LA) (DOE, 2008a, pg. ES-9], DOE states: "Typically, when two or more models exist for the same phenomena and data, the more conservative one from a total-system perspective has been chosen for implementation."

For nearly 20 years, the Electric Power Research Institute (EPRI) has been reviewing the U.S. Department of Energy's development of the proposed geologic repository for disposal of high-level radioactive waste and spent nuclear fuel at Yucca Mountain, Nevada. Independent analyses and data collection conducted by EPRI suggest that there are many issues with respect to the current DOE design and analyses, as presented in the License Application, that may result in unnecessary occupational health hazards to workers in the nuclear industry and other related industries.

In its Yucca Mountain Review Plan (NRC, 2003), NRC states:

> Consideration of radiological risk in the design and construction of the repository and the limitation of such risk is also consistent with a commitment to the 'As Low as Reasonably Achievable' (ALARA) principles of Regulatory Guide 8.8, as is called for in 10 CFR Part 20 and Section 2.1.1.8 of NUREG 1804, the Yucca Mountain Review Plan (YMRP).

Thus, NRC is stating it will review DOE's Yucca Mountain license application with consideration of ALARA principles. In this report, EPRI has interpreted NRC (2003) to mean ALARA principles should be considered for the *entire* spent fuel waste management process – from storage of commercial spent nuclear fuel (CSNF) at the reactor sites, loading and transfer of CSNF at the reactor sites, transportation of the CSNF to receipt, handling, and disposal of the CSNF at Yucca Mountain. EPRI has considered both radiological and non-radiological occupational health and safety risks during reactor-site storage, CSNF transfer and loading, CSNF transportation, CSNF management at Yucca Mountain, and construction of appropriate CSNF management facilities at the reactor sites and at Yucca Mountain.

The purpose of this report is to identify those issues and provide semi-quantitative estimates of the "unnecessary" occupational health risks that may result from the DOE Yucca Mountain analyses and repository design such that the proposed analyses and designs are not consistent with ALARA principles. While EPRI recognizes there could be additional, "unnecessary" health hazards to the public due to DOE's analysis and design, public health hazards are not assessed quantitatively in this report. Except on a limited basis, neither will this report quantitatively estimate economic consequences of unnecessary or inappropriate elements of the DOE design or analyses.

For purposes of this report, the term “unnecessary” is intended to mean the additional risk that may be incurred by performing an activity in the manner proposed by DOE versus the more limited amount of risk that may be incurred by performing the activity in some alternative manner that EPRI considers to be more consistent with the principles of ALARA. The difference between the two levels of risk is considered by EPRI to be “unnecessary.” EPRI recognizes that there are a certain amount of hazards and risks associated with all such activities and that it is impossible to reduce such hazards and risks to zero.

In this report, the terms “risk”, “hazard”, “impact”, “consequence”, among others are used in their most general sense and interchangeably to denote the undesirable outcome or effect that results from an action, assumption, or decision made by DOE in its approach to the design, assessment, and operation of Yucca Mountain. EPRI recognizes that these terms also have more precise technical meanings.

As in other EPRI reports, the intent of this report is not to present worst-case analyses, but rather to adhere to the intent of the EPA’s proposed regulatory structure in 40 CFR 197 (EPA, 2005), which is to provide more realistic analyses:

> Overly conservative assumptions made in developing performance scenarios can bias the analyses in the direction of unrealistically extreme situations, which in reality may be highly improbable, and can deflect attention from questions critical to developing an adequate understanding of the expected features, events, and processes (“Assumptions, Conservatisms, and Uncertainties in Yucca Mountain Performance Assessments,” Sections 11 and 12, July 2005, Docket No. OAR-2005-0083-0085). The reasonable expectation approach focuses attention on understanding the uncertainties in projecting disposal system performance so that regulatory decision making will be done with a full understanding of the uncertainties involved. Thus, realistic analyses are preferred over conservative and bounding assumptions, to the extent practical. (40 CFR 197: EPA, 2005)

According to 40 CFR 197.14, “reasonable expectation”:

- “Requires less than absolute proof because absolute proof is impossible to attain…”
- “Accounts for the inherently greater uncertainties in making long-term projections…”
- “Does not exclude important parameters from assessments and analyses simply because they are difficult to precisely quantify…”
- “Focuses performance assessments and analyses upon the full range of defensible and reasonable parameter distributions rather than only upon extreme physical situations and parameter values”

While some conservatism in the face of uncertainty is warranted, especially given the proposed one million year compliance period for repository performance, repeated application of overly conservative assumptions and estimates in performance assessment will likely result in overly designed facilities in order to provide excess performance margins for the protection of the health of hypothetical future lives at the expense of present day workers and public. Overly conservative and unrealistic assessment of repository performance is not a risk-neutral endeavor. Each additional activity undertaken by DOE and its contractors during construction, operation, and closure of the repository carries with it finite levels of risk to the workers that must carry out those activities. Moreover, assumptions integral to the DOE proposed approach to the repository also have serious consequences for the utilities that currently manage the spent nuclear fuel onsite in wet and/or dry storage configurations.

DOE's cleanup efforts under its Environmental Management program have been repeatedly criticized for what has been termed "the unacknowledged transfer of risk" (Young and Wood, 2001; Church, 2001) in which conservative assumptions drive costly remedial actions that impose unjustified risks of fatalities and injuries to workers and the public during construction and transportation.

Workers, including those at utility sites, are likely to bear the greatest burden associated with such risk transfer each time DOE chooses overly conservative options in its repository design, analyses, and operational planning.

Workers are likely to bear the greatest burden associated with such unintended and unjustified transfers of risk each time DOE chooses overly conservative options in its repository design, analyses, and operational planning. Unjustified and unnecessary elements of the DOE license application represent an unfair and unjustified transfer of risk from hypothetical future lives to existing nuclear industry and utility workers, as well as present day members of the public.

The purpose of this report is bring attention to elements of the DOE total-system performance assessment and proposed approach to the repository design, construction, and operation, as presented in the 2008 license application and supporting documents, that could result in additional, non-trivial risk burdens for present day workers both in terms of radiological and non-radiological risks, and to provided quantitative estimates of those risks, where possible.

1.1 Issues and Potential Consequences for Occupational Health

The issues and potential unnecessary occupational health hazards are summarized in the following subsections. Each of these issues and their effect on occupational health risks are discussed in more detail in the following chapters.

1.1.1 Some Dual-purpose Canisters are Suitable for Direct Disposal

EPRI analyses suggest that at least some of the existing dual-purpose canisters (DPCs) used by the nuclear industry could be safely transported, aged, and disposed of at Yucca Mountain. Currently licensed DPCs hold approximately 1.14 to 1.55 times as much SNF as do TADs. Thus, using TADs instead of DPCs will result in 1.14 to 1.55 times as many canisters being loaded at nuclear utility sites, transported to Yucca Mountain, potentially aged, and eventually emplaced in the repository.

Potential impact on occupational health:
The DOE decision to not consider direct disposal of DPCs in its License Application imposes significant unnecessary occupational health risks on workers associated with the operations needed to open the loaded DPCs, transfer the CSNF to a TAD canister, manage the empty DPCs as low-level radioactive waste (LLW), and close the newly loaded TAD. Also significant would be the additional occupation risks borne by workers due to the need for additional loading TAD canisters arising from the limited capacity of the TAD versus larger capacity DPCs.

1.1.2 The Size of the Proposed Transportation, Aging, and Disposal Canisters is Smaller than is Necessary

DOE has proposed the use of transportation, aging, and disposal canisters (TADs) such that the utilities would load commercial spent nuclear fuel (CSNF) into the TAD canisters at the reactors, and with appropriate transportation, aging, and disposal overpacks, the TAD canisters would not need to be reopened after closure at the reactor sites. DOE also proposed to use TAD canisters for CSNF it will receive at Yucca Mountain from the utilities that would arrive in shipping containers other than TADs. The proposed capacity of the TADs is 21 pressurized water reactor (PWR) assemblies or 44 boiling water reactor (BWR) assemblies. Assuming DOE and the utilities reach agreement on the use of TADs at reactor sites, the sizes of the TADs are smaller than is necessary to reliably meet EPA and NRC regulatory performance criteria. EPRI analyses suggest that TADs could be up to 1.55 times larger without impinging on overall repository performance or exceeding thermal design limits. Thus, using TADs instead of DPCs will result in up to 1.55 times as many canisters being loaded at nuclear utility sites, transported to Yucca Mountain, potentially aged, and then disposed.

Potential impact on occupational health:
Using the proposed 21P/44B TAD size compared to use of a larger TAD, with a capacity that is similar to larger capacity DPCs currently in use for on-site dry storage, will result in additional unnecessary radiological and non-radiological risks borne by workers at utility sites, at Yucca Mountain itself, and in the transportation sector. These impacts result from the need for additional activities associated with canister loading, transport, and handling at Yucca Mountain. Each additional waste package will require excavation of an additional length of emplacement drift. Additional installation of drift hardware (invert, pallet, drip shield) and subsurface infrastructure (rock bolts, tunnel (mesh) liner), along with additional person-hours of labor associated with all aspects of handling, maintenance, inspection, and emplacement. In additional, manufacturing of additional repository system components for waste packages and developed drift components, will incur additional occupational risk during their manufacture and transport.

1.1.3 DOE Underestimated the Amount of Commercial Spent Nuclear Fuel Arriving at Yucca Mountain not in TADs

Even assuming the use of TADs, DOE has underestimated the amount of commercial spent nuclear fuel (CSNF) that would be shipped to Yucca Mountain in non-TADs. While DOE estimates a base case of 10% and a maximum of 25% of the CSNF would be shipped in non-TADs, EPRI estimates that more than 25% of the CSNF will already have been placed in non-TAD containers. At present, the DOE and utilities have not entered into specific agreements regarding the use of TADs for Yucca Mountain disposal, yet the proposed action does not specifically provide for CSNF acceptance in any form other than in TADs or as bare fuel.

Potential impact on occupational health:
Since DOE has underestimated the amount of CSNF that will be stored in canisters other than TADs at the reactor sites (mostly in DPCs), it may be necessary for workers to open and unload even more DPCs than discussed in Section 1.1.1. The use of additional TADs and the potential need to repackage CSNF already in DPCs at Yucca Mountain, and potentially at the utility sites, will cause increases in potential occupational hazards with respect to the reopening and unloading of existing DPCs, CSNF transfer from the DPCs into TADs, TAD closure, and

preparation of the TAD and its transportation overpack for shipment of CSNF to Yucca Mountain. In addition, there would be additional handling of CSNF in more TADs (relative to the number of DPCs due to the TADs' lower CSNF capacities) at Yucca Mountain. By requiring that only a fraction of the CSNF that will exist in DPCs or other storage canisters can be shipped to Yucca Mountain without repackaging into TADs, there will be increased occupational risks associated with additional handling of CSNF in DPCs including radiological and non-radiological risks.

1.1.4 The Probability of Igneous Activity within the Repository Footprint has been Overestimated

EPRI has determined that the probability of an igneous event intersecting the Yucca Mountain repository is less than 10^{-8} per year. As such, potential consequences of igneous activity need not be presented in DOE's license application per the draft 40 CFR 197 Yucca Mountain regulation. Furthermore, EPRI has determined that DOE's estimates of consequences due to igneous eruption and intrusion scenarios have been overstated.

Potential impact on occupational health:
Including igneous consequence analysis in the Yucca Mountain licensing proceedings could cause unnecessary delays in the licensing proceedings by deflecting attention from questions critical to developing an adequate understanding of the expected features, events, and processes. This could cause nuclear utilities to have to load and store additional spent nuclear fuel at the reactor sites, leading to additional radiation dose to both workers and the public nearby to the spent fuel storage facilities. In addition, workers involved with loading and transferring spent fuel storage casks at the utility sites would be exposed to additional, non-radioactive hazards involved with potential accidents leading to worker injury.

1.1.5 Drip Shields are Unnecessary

There are several conservatisms in DOE's analyses of post-closure performance that have led DOE to unnecessarily include drip shields in its repository design.

- Overestimation of the amount of net infiltration thereby incorrectly indicating a larger benefit of the use of a drip shield than is actually the case;
- Overestimation of the fraction of the repository experiencing seepage into the open drifts, having the same effect as overestimation of net infiltration;
- Overestimation of seismic energy and rockfall. This leads DOE to the conclusion that drip shields would provide significant protection from rockfall;
- Overestimation of damage to the TADs due to seismic and rockfall events. This also leads to the incorrect conclusion that drip shields would provide additional protection from damage of the waste packages;
- Overestimation of the rate at which Alloy 22 will degrade. This, in turn, gives greater performance credit to the drip shields than is warranted. This could lead to additional, unnecessary regulatory scrutiny that could delay the licensing process;
- Cladding performance has been neglected. EPRI analyses indicate that including cladding performance would provide an additional barrier to the release of radionuclides from the waste form. This would also reduce the need for a drip shield;

- DOE notes that it typically uses the more conservative of two or more conceptual models. Some of these conservatisms could also result in the apparent need for drip shields.

Potential impact on occupational health:
The construction, transportation, and installation of drip shields would cause unnecessary, radiological and non-radiological occupational health hazards. Mining of titanium, conversion to metal, and manufacture of the drip shields would cause unnecessary industrial hazards to the relevant workers and will put pressure on available titanium resources. Installation of the drip shields would also impose unnecessary risks to Yucca Mountain workers.

1.1.6 The Surface Facilities have been Overdesigned to Withstand Seismic Ground Motion

DOE has assessed the risk of seismic ground motion during the pre-closure period. While it is certainly necessary to design systems, structures, and components to withstand this risk, EPRI believes DOE's surface facility is overdesigned for this risk. This has led to an unnecessarily large, robust surface facility structures and elements.

Potential impact on occupational health:
Additional health risks to workers and the public caused by the construction of over-designed surface and sub-surface facilities would be caused by, for example, transportation and use of additional construction materials and additional, unnecessary construction activities.

1.1.7 DOE Overestimated the Seismic Energy that is Possible During the Post-closure Period

EPRI contends that DOE's estimates of seismic energy risk at Yucca Mountain are overstated – especially for the long recurrence interval seismic events. Because DOE has overestimated seismic energy, it has also overestimated the amount and timing of rockfall (especially during the time period shortly after repository closure). This has led to an overestimate of dose to the public in DOE's analyses, especially for early times after repository closure.

Potential impact on occupational health:
This could also cause a delay in the availability of the Yucca Mountain repository if, for example, DOE needs to perform additional, unnecessary construction tasks to accommodate DOE's overestimate of seismic energy. Furthermore, EPRI feels that one of the reasons DOE has specified a very robust TAD design is to mitigate damage to the TAD overpack that could be caused by the seismic energy overestimates. Additional delays in the ability to move CSNF from reactor sites to the Yucca Mountain repository could be caused by the need to develop, license, construct, load, and dispose of unnecessarily robust TAD canisters and overpacks. Delays in the ability to move CSNF to Yucca Mountain could cause both occupational and public radiological and non-radiological health hazards.

1.1.8 Co-disposal versus TAD Waste Package Design and/or Analysis Caused the Peak Dose to be Driven by Co-disposal Waste Packages

It appears that DOE's TSPA indicates the first peak in post-closure dose is due primarily to the relatively early failure of the co-disposal waste packages compared to the now very robust TAD waste packages for CSNF. The first peak is roughly the same magnitude as the peak due primarily to TAD failure many hundreds of thousands of years in the future. There are also conservatisms in how DOE calculates the peak dose for the co-disposal waste packages.

Potential impact on occupational health:
The fact that DOE has estimated the peak dose due to co-disposal waste packages is roughly the same as from the TADs containing CSNF may cause unnecessary regulatory scrutiny, thereby leading to potential licensing delays. Occupational health impacts due to delays in opening the repository have been discussed earlier.

1.1.9 The Spacing between Disposal Drifts is Unnecessarily Large

DOE's drift center-to-center spacing requirement of 81 meters is based on conservative estimates of temperature in the rock pillars over time, as well as the artificially imposed requirement of keeping some of the rock pillar below boiling temperatures at all times. The result of the unnecessarily large drift spacing is that more rock will need to be excavated, and more rock supports will need to be installed than is actually necessary.

Potential impact on occupational health:
Excavation of additional rock and installation of additional rock supports will increase both the radiological and non-radiological hazard to workers excavating the drifts and installing the rock support, as well as occupational and public health hazards due to the transportation of extra rock support materials;

1.1.10 The Waste Handling Facility Throughput DOE proposes is Insufficient to Process the CSNF that will be Shipped to Yucca Mountain not in TADs

As discussed above, EPRI concludes there will be more CSNF shipped to Yucca Mountain that would need to be processed in DOE's Wet Handling Facility (WHF) than DOE is planning in its Proposed Action. Either DOE will need to construct additional WHFs or it will take longer to process the larger amount of CSNF in one WHF.

Potential impact on occupational health:
If all 63,000 MTHM of CSNF is to be processed in 24 years as DOE proposes, additional WHFs will have to be constructed, with the concomitant increase in occupational health risks due to material fabrication and transportation, and construction activities. Additional WHF construction will likely lead to a delay in the ability to transfer CNSF from reactor sites to Yucca Mountain. Alternatively, if just one WHF is constructed, the it will require additional processing time, which could cause nuclear utilities to have to load and store additional spent nuclear fuel at the reactor sites, leading to additional radiation dose to both workers and the public nearby to the spent fuel storage facilities. In addition, workers involved with loading and transferring spent fuel storage casks at the utility sites would be exposed to additional, non-radioactive hazards involved with potential accidents leading to worker injury.

1.1.11 Conservatisms in DOE Analyses Led to an Overestimate of Post-closure Dose

EPRI has determined that DOE's TSPA has incorporated many conservatisms that have led DOE to overestimate dose rates to the RMEI during the post-closure period. These many conservatisms cannot simply be considered independently, since many conservatisms compound with others, so that the net effect is greater than each taken individually.

Potential impact on occupational health:
Because DOE's multiple conservatisms cause DOE to overestimate dose rates to the RMEI, the repository system design may be more robust than a repository design based on a different design based on more reasonable assumptions and data inputs to DOE's dose assessment calculations. Secondarily, the loss of margin below the draft EPA and NRC dose limits has the potential to increase the licensing process. Either of these causes could lead to a delay in the availability of Yucca Mountain. Any delay in the licensing, construction, and operation of the repository places additional radiological and non-radiological risk burdens on workers at the utility sites due to the need to construct additional ISFSI capacity; to extend and/or expand inspection and maintenance programs for existing ISFSI facilities at operating plants.

1.2 Approach

EPRI's approach in developing the analyses in this report was to utilize, as possible and appropriate, cautious but realistic assumptions in the performance of its various analyses and investigations, as recommended by the National Academy of Sciences in its *Technical Bases for Yucca Mountain Standards* report (NAS, 2005). For example:

- Occupational risk is considered only for involved workers, although it is recognized that each additional unit of activity requires the support of professionals and other ancillary staff that are not directly exposed to the hazards of the work site but still incur risk associated with office settings and travel to and from work. These additional workers are typically referred to a "non-involved workers."
- Whenever possible and where deemed appropriate, EPRI utilized DOE data and estimates obtained from the various Yucca Mountain related documents such as the Environmental Impact Statements, the License Application itself, and supporting documents and calculation packages. This was done in order for EPRI to be able to make direct comparison between its assessment of worker risk and the risk calculations contained in the DOE documents. In the event that the DOE data and estimates were not available or are did not provide enough supporting detail to allow for derivative analysis, EPRI used publicly available data from the U.S. Nuclear Regulatory Commission (NRC), Bureau of Labor Statistics (BLS), and other citable sources.
- DOE performed a detailed assessment of impacts to workers at nuclear power plants sites and DOE sites in its analysis of the No Action Alternative for the Yucca Mountain EIS, as supplemented. EPRI relied on some of the at-reactor worker impacts utilized by DOE in its No Action Alternative analysis. When available, EPRI has also identified other citable sources of data associated with worker impacts at nuclear power plant sites.
- Collective occupation dose is the primary metric used in this report for tracking radiological risk burdens as it provides a convenient means for tracking such risks to workers without the need to make assumptions about how a company, utility, or DOE contractor divides that burden among its workforce. While the use of collective dose has important limitations, here it is used as exclusively an accounting tool and not for causally linking specific health effects to low exposures.
- Radiological hazards to workers during transport are evaluated for accident free transport only. Radiological exposure associated with transportation accidents is not considered.
- Transportation accidents are considered for evaluating non-radiological risks to workers.

- Non-radiological hazards are primarily tracked via the standard Bureau of Labor Statistics categorization of total recordable cases (TRC), and lost workday cases (LWC). and fatalities and are typically indexed to full-time equivalent.worker years (FTE).
  - Total Recordable cases include Recordable cases include work-related injuries and illnesses that result in one or more of the following: death, loss of consciousness, days away from work, restricted work activity or job transfer, medical treatment (beyond first aid), significant work-related injuries or illnesses that are diagnosed by a physician or other licensed heath care professional
  - Lost work-time cases include all cases involving days away from work, or days of restricted work activity, or both.
  - Fatalities include all cases of work related deaths.
  - Non-radiological health and safety data are presented either as a rate (number of cases per X number of FTE) or as total number of cases.
  - A full-time equivalent worker year is equivalent to 2,000 work hours, i.e., the typical number of hours for a typical worker year comprised of 8 hours per day, 50 weeks per year.

The occupational health impacts resulting from the approaches taken by DOE in its Yucca Mountain design, analyses and operations are estimated in Appendices B and C of this report, with supporting data presented in Appendices A, D, and E.. Most estimates are provided on a generic basis using the best available data and what are deemed to be reasonable assumptions. These estimates are then used to calculate overall impacts to the extent data and assumptions allow. However, in some cases, the estimated impacts may be provided for "unit" increments of:

1. Time (for example, the impact due to a delay of opening the repository by one year);
2. Individual operational steps (for example, the occupational impact of loading one additional TAD canister);
3. Length of access or disposal drifts (for example, the occupational impact of having to excavate and develop an extra one meter of drift); or
4. Facility construction units, such as the construction of one additional Wet Handling Facility or the use of additional concrete and building materials.

These unit values are used, when possible, to estimate the occupational health effects for each one of the issues in the following chapters.

2
SOME DUAL-PURPOSE CANISTERS (DPCS) ARE SUITABLE FOR DIRECT DISPOSAL

2.1 Technical Bases

The License Application states that DOE has rejected the idea of directly disposing of *any* DPCs in favor of repackaging the CSNF into TADs prior to disposal:

> DPCs are currently used by several utilities to store and potentially ship commercial SNF. Currently licensed DPCs have not been shown to be suitable for disposal purposes. However, although not currently acceptable under the provisions of 10 CFR Part 961, the DOE may choose to receive DPCs at the repository and repackage the commercial SNF into a TAD canister for disposal after the execution of mutually agreeable amendments to the utilities disposal contract. (DOE 2008b, Section 1.5.1.1.1.2.1.2)

DOE also defines a “disposable canister” as:

> A metal vessel for commercial and DOE spent nuclear fuel assemblies ... or solidified high-level radioactive waste suitable for storage, shipping, and disposal. At the repository, DOE would remove the disposable canister from the transportation cask and place it in a waste package. There are a number of types of disposal canisters, including DOE standard canisters, multicanister overpacks, naval spent nuclear fuel canisters, and TAD canisters. (DOE 2008d, Section 2.1.1)

EPRI evaluated the possibility of the larger DPCs meeting DOE’s criterion for a “disposable canister” against several criteria (EPRI, 2008a):

- Size -- to determine if the inner DPC canister plus a modified disposal overpack (modified to fit the DPC canister, but otherwise dimensionally consistent with the proposed TAD design) will fit inside the proposed disposal drift diameter, and still allow room for installation of the invert, pedestal, drip shield, and rock support;
- Rock wall temperature -- to determine if direct disposal of DPCs will cause rock wall temperatures to exceed ~200°C. This temperature limit is a reasonable upper bound that would prevent significant rock expansion leading to potentially significant rock spallation. However, previous EPRI analysis suggests this temperature limit could be increased to ~225°C (EPRI, 2006a), if necessary.
- Seismicity and rockfall – to determine if there are any special issues with respect to the ability of DPCs to withstand anticipated seismic and rockfall events;
- Pillar dry-out – to determine if the water saturation in some of the rock between the disposal drifts remains above zero, thereby allowing passage of groundwater infiltrating from above the repository to below the repository. While beneficial, EPRI contends that it is not necessary to maintain water saturation in the pillar above zero at all times (EPRI, 2006a; 2007a);

- Criticality – to determine if DPCs in appropriate disposal overpacks will remain sub-critical during the post-closure period, or if critical for some scenarios, whether the canisters are likely to become prompt critical (EPRI, 2007b; 2008a); and
- Long-term dose to the RMEI (reasonably maximally exposed individual) – to compare the peak RMEI dose in the post-closure period due to the disposal of CSNF in DPCs with disposal overpacks with that due to the disposal of TADs.

EPRI (2008a) and EPRI (2007b) find there are no known technical barriers to direct disposal of at least some of the DPCs. Peak temperatures at the rock wall and in the rock pillars will not exceed values to cause excessive rock spalling and pillar dry-out, respectively:

> Direct DPC disposal was examined to determine if there would be any significant issues relative to thermal effects, thermal-mechanical effects, corrosion, TSPA of the nominal repository evolution scenario and credible alternative repository evolution scenarios, as well as criticality. It is concluded that there are very small differences in performance of DPCs in the post-closure period compared to performance of TADs. Criticality is also extremely unlikely for both TADs and DPCs. No obstacles have been identified that would preclude the use of DPCs for disposal of commercial spent nuclear fuel (CSNF) in a geologic repository at Yucca Mountain. …
>
> Both TADs and a significant portion of the DPCs that will exist at the time of TAD availability are disposable. For the sizeable inventory of CSNF already safely sealed in DPCs, EPRI believes that … a substantial inventory of dual-purpose casks, which are designed for storage and transport, could be certified for disposal at Yucca Mountain based on performance based criteria.

Therefore, EPRI argues that at least some of the DPCs anticipated to be in existence at the time DOE is ready to accept CSNF at Yucca Mountain *can* be disposed of directly by inserting them inside an appropriate Alloy 22 outer canister.

2.2 Occupational Health Risk Impacts

The DOE decision to not consider direct disposal of any DPCs in its License Application imposes significant unnecessary occupational health risks on workers associated with the operations needed to open the loaded DPCs, transfer the CSNF to a TAD canister, manage the empty DPCs as low-level radioactive waste (LLW), and close the newly loaded TAD. Also, significant additional occupation risks would be borne by workers due to the need for additional loading TAD canisters arising from the limited capacity of the TAD versus larger capacity DPCs.

The occupational health impacts caused by the need to transfer CSNF from the DPC into TADs, presumably at Yucca Mountain, are described in detail in Appendices B and C. Some key impacts are summarized in Table 2-1. For DPC systems transported to Yucca Mountain and unloaded, rather than being placed in waste packages for direct disposal, a net additional worker dose of 135 person-mrem per package (260 person-rem – 125 person rem from Table B-6) is incurred (Table B-8). Accordingly, this same dose also represents the potential dose avoided per canister if DPCs or other existing, loaded canister systems were qualified by DOE for direct disposal.

Table 2 - 1
Net Occupational Doses Associated with Unloading and Disposal of DPCs

DPC scenario	Number of DPCs for Receipt at Yucca Mountain	Worker Dose Associated with DPC Unloading (person-rem)	Worker Dose Associated with DPC Waste Management (person-rem)
DOE baseline	307	80	14
DOE high estimate	966	250	43
EPRI high estimate	2375	620	110

Likewise, each emptied DPC (or other canister) will need to be managed as LLW, incurring estimated additional doses to workers of 0.045 person-rem for each DPC discarded. Thus, the dose in Table 2-1 represents both the estimated dose to workers associated with LLW management activities under the DOE proposed operational approach and the dose that could be avoided if DPCs or other existing, loaded canister systems were employed for direct disposal in Yucca Mountain.

The additional handling steps associated with unloading and disposing of DPCs also pose additional potentially unnecessary occupational risk to workers at Yucca Mountain (or reactor sites should unloading operations be required prior to shipment). EPRI was not able to develop specific estimates for these impacts, but the DOE considers the following industrial injury and fatality rates for workers at Yucca Mountain during operations:

- TRC 1.4 per 100 FTE
- LWC 0.58 per 100 FTE
- Fatalities 0.55 per 100,000 FTE

3
TAD CANISTER CAPACITY IS SMALLER THAN NECESSARY FOR DISPOSAL

EPRI analyses conclude that the Transportation, Aging, and Disposal (TAD) canisters and disposal overpacks are smaller than could be used for disposal at Yucca Mountain. Thus, the sizes of the TADs are smaller than necessary. As discussed in EPRI (2008a) and summarized in Section 2.1 of this report, EPRI finds that many of the existing dual-purpose canisters (DPCs) used by the nuclear industry could be safely transported, aged, and disposed of at Yucca Mountain. Currently licensed DPCs hold approximately 1.14 to 1.55 times as much spent nuclear fuel as do the proposed TADs. Thus, using the proposed TAD size instead of DPCs or larger capacity TADs will result in a larger number of canisters being loaded at nuclear utility sites, transported to Yucca Mountain, potentially aged, and then disposed of in the repository.

Section 3.1 makes the argument that TADs capacities could be at least as large as DPCs that have a capacity of 1.5 times that of the DOE-proposed TAD capacities. Section 3.2 discusses the avoidable occupational health risks by increasing the capacity of the TADs by a factor of 1.5.

3.1 Technical Bases

DOE proposes to use TADs for the transportation, aging, and disposal of CSNF (DOE, 2008b). The proposed TAD canisters would hold 21 PWR assemblies or 44 BWR assemblies. This TAD size is termed a "21P/44B". While EPRI agrees that TADs of this size can be safely transported, aged, and disposed of at Yucca Mountain, it is also possible to use larger waste packages (including both the inner canisters and the relevant overpacks for transportation, aging, or disposal).

U.S. nuclear utilities are currently using a variety of CSNF dry storage systems at their reactor sites. The earliest dry storage systems were designed for storage-only operations; later designs are almost exclusively "dual-purpose" canisters – designed for both dry storage and transportation. However, most DPCs are currently certified for storage only. Many of the utilities using the storage-only systems have or are in the process of submitting license applications to the NRC to certify these systems for transport. While a handful of the earliest storage-only systems are smaller than the 21P/44B TAD capacity, the majority of storage-only and DPCs are larger than 21P/44B.

Section 2.1 summarizes EPRI's conclusion that some DPCs could be considered "disposable canisters". EPRI considered a DPC capacity 1.5 times as large as the DOE-proposed 21P/44B TAD. Given that EPRI concludes some of the larger DPCs can be directly disposed of (EPRI, 2008a), EPRI argues that larger TAD capacities could have been selected by DOE based on findings from EPRI's evaluation of larger DPCs for direct disposal, which apply to large TAD designs as well.

EPRI evaluated the possibility of direct disposal of the larger DPCs against several criteria (EPRI, 2008a):

- Size -- to determine if the inner DPC canister plus a modified disposal overpack (modified to fit the DPC canister, but otherwise dimensionally consistent with the proposed TAD design) will fit inside the proposed disposal drift diameter, and still allow room for installation of the invert, pedestal, drip shield, and rock support;
- Rock wall temperature -- to determine if direct disposal of DPCs will cause rock wall temperatures to exceed ~200°C. This temperature limit is a reasonable upper bound that would prevent significant rock expansion leading to potentially significant rock spallation. However, previous EPRI analysis suggests this temperature limit could be increased to ~225°C (EPRI, 2006a), if necessary.
- Seismicity and rockfall – to determine if there are any special issues with respect to the ability of DPCs to withstand anticipated seismic and rockfall events;
- Pillar dry-out – to determine if the water saturation in some of the rock between the disposal drifts remains above zero, thereby allowing passage of groundwater infiltrating from above the repository to below the repository. While beneficial, EPRI contends that it is not necessary to maintain water saturation in the pillar above zero at all times (EPRI, 2006a; 2007a);
- Criticality – to determine if DPCs in appropriate disposal overpacks will remain sub-critical during the post-closure period, or if critical for some scenarios, whether the canisters are likely to become prompt critical (EPRI, 2007b; 2008a); and
- Long-term dose to the RMEI (reasonably maximally exposed individual) – to compare the peak RMEI dose in the post-closure period due to the disposal of CSNF in DPCs with disposal overpacks with that due to the disposal of TADs.

EPRI (2008a) and EPRI (2007b) find there are no known technical barriers to direct disposal of at least some of the DPCs. Peak temperatures at the rock wall and in the rock pillars will not exceed values to cause excessive rock spalling and pillar dry-out, respectively.

3.2 Potential Impacts of Using a Smaller TAD

Using the proposed 21P/44B TAD size compared to use of a larger TAD, with a capacity that is similar to larger capacity DPCs currently in use for on-site dry storage, will result in additional unnecessary radiological and non-radiological risks borne by workers at utility sites, at Yucca Mountain itself, and in the transportation sector. These impacts result from the need for additional activities associated with canister loading, transport, and handling at Yucca Mountain.. Each additional waste package will require excavation of an additional length of emplacement drift. Additional installation of drift hardware (invert, pallet, drip shield) and subsurface infrastructure (rock bolts, tunnel (mesh) liner), along with additional person-hours of labor associated with all aspects of handling, maintenance, inspection, and emplacement. Furthermore, manufacturing of additional repository system components for waste packages and developed drift components, will incur additional occupational risk during their manufacture and transport.

EPRI evaluated the potential occupational health and safety impacts associated with DOE's decision to exclusively use the proposed 21P/44B TAD rather than use of larger TAD designs. For the reactor site and transportation activities, these effects are the same as for DOE's decision to not consider direct disposal of larger DPCs. This is because it is assumed that the transfer of CSNF from DPCs to TADs would occur at Yucca Mountain, per DOE's Proposed Action.

The evaluation considered here uses two alternative scenarios, EPRI Case 1 and EPRI Case 2. Case 1 assumes that larger (32-PWR/68-BWR) TADs are deployed for loading of fuel at reactor sites, leading to concomitant reductions in loading operations, shipments, handling, and drift length. Case 2 extends Case 1 further to exclude the exclusive truck shipments from seven reactor sites that are assumed in DOE's baseline estimate. The resulting occupational impacts are summarized in Table 3-1 below.

The basis for these estimates are provided in Appendices A, B. and C for quantities of required canisters/casks, radiological impacts, and non-radiological impacts respectively.

Table 3 - 1
Radiological and Non-Radiological Impacts of Using TADs that are Smaller than Necessary

Affected Worker Population	EPRI Scenario for Comparison	Source of Impact	Additional Cumulative Dose (person-rem)	Additional Injuries and Fatalities
Reactor sites	Case 1	21P/44B TAD capacity results in additional canister loading	2,028 Table B-2	19 TRC 13 LWC 0.04 fatalities
	Case 2	21P/44B TAD capacity and assumption of 7 nuclear plants shipping by truck results in additional package loading	2,813 Table B-2	31 TRC 21 LWC 0.07 fatalities
Transportation	Case 1	21P/44B TAD capacity results in additional shipments of CSNF to the repository	1,174 Table B-5	Rail accident: 1.15×10^{-8} fatality/ railcar-km For shipments involving 3 CSNF casks (8 railcars total), the fatality rate was estimated to be 9.20×10^{-8} accidents/train-km
	Case 2	21P/44B TAD capacity and assumption of 7 nuclear plants shipping by truck results in additional shipments of canisters	1,783 Table B-5	Truck accident 5.34E-07 accidents per truck km 1. 55E-08 fatalities per truck km

Table 3-1 (continued)

Affected Worker Population	EPRI Scenario for Comparison	Source of Impact	Additional Cumulative Dose (person-rem)	Additional Cumulative Dose (person-rem)
Yucca Mountain operations	Case 1	21P/44B TAD capacity results in additional canisters for receipt and handling	701 Table B-7	1.4 TRC per 100 FTEs 0.58 LWC per 100 FTE 0.55 fatalities per 100,000 FTW worker years
	Case 2	21P/44B TAD capacity and assumption of 7 nuclear plants shipping via truck casks results in additional packages for receipt and handling	1,792 Table B-7	1.4 TRC per 100 FTEs 0.58 LWC per 100 FTE 0.55 fatalities per 100,000 FTW worker years
Yucca Mountain subsurface construction	Case 1	Drift excavation to accommodate additional CSNF waste packages	155	18 TRC 7.7 LWC 0.0049 fatalities
	Case 2	Drift excavation to accommodate additional CSNF waste packages	166	19 TRC 8.2 LWC 0.0052 fatalities

Other Health and Economic Impacts

Additional Radiological Health Impacts to Workers at Reactor Sites Associated with Unloading Storage-Only Dry Storage Systems

While the YMSEIS did not calculate the worker dose associated with unloading CSNF in dry storage at reactor sites for repackaging prior to shipment to Yucca Mountain, it is possible that some of these packages would be unloaded at reactor sites. EPRI assumes that industry workers would incur a dose of 260 person-mrem per package unloaded, as identified in B-1. If storage only casks must be unloaded, this will result in an estimated worker dose of 83 person-rem. If dual-purpose metal casks must be unloaded at reactor sites, the estimated worker dose would be 35 person-rem. If DPCs and storage-only canisters are unloaded at reactor sites for repackaging, the estimated worker dose would be 617 person-rem. (Table B-4)

Radiological Health Impacts to the Public During TAD Transportation from the Reactor Sites to Yucca Mountain :

Incident-Free Transportation Radiation Doses:

- Rail: 800 person-rem
- Truck: 350 person-rem

The use of higher capacity TAD designs as well as the shipment of CSNF in higher capacity TAD designs from sites identified by DOE as truck sites, would result in fewer packages being shipped. This would result in a proportional decrease in the incident-free dose to the public similar to the reduction in worker dose during transport discussed in Appendix B.

Non-radiological Impacts to the Public during TAD and Ancillary Equipment Transport to Reactor Sites

The YMSEIS assumed that approximately 6,500 empty TAD canisters would be shipped to commercial reactor sites by truck under the 70,000 MTU repository scenario. In addition to the shipment of TADs, approximately 4,900 kits of ancillary equipment needed for loading at reactor sites would also be shipped. DOE assumed that a total of 1.2 traffic fatalities would result from these shipments and 0.23 fatalities from vehicle emissions (assuming a shipping distance of 3,000 kilometers per shipment). (DOE 2008a, Section 6.2.1). If higher capacity TAD canisters were used to load CSNF as described by EPRI Case 1 or EPRI Case 2, a fewer number of TAD canisters and ancillary equipment would need to be transported resulting in a smaller number of vehicle fatalities and vehicle emission fatalities,

Economic Impacts

Increase in costs associated with DOE's proposal to use 21P/44B TADs compared to EPRI Case 1:

▪ At reactor loading costs	$0.38 billion
▪ Transport costs	$0.33 billion
▪ Disposal costs (TAD canisters and waste packages	$3.14 billion
▪ Total potential cost impacts:	$3.85 billion

Increase in costs associated with DOE's proposal to use 21P/44B TADs compared to EPRI Case 1:

▪ At reactor loading costs	$0.44 billion
▪ Transport costs	$0.41 billion
▪ Disposal costs (TAD canisters and waste packages)	$3.33 billion
▪ Total potential cost impact	$4.18 billion

3.4 Summary of Impacts

Using the proposed 21P/44B TAD size compared to use of a larger TAD will result in increases in radiological and non-radiological risks borne by workers at utility sites, at Yucca Mountain itself, and in the transportation sector. These impacts result from the need for additional activities associated with canister loading, transport, and handling at Yucca Mountain.

As shown in Table 3-1, comparing DOE's proposed 21P/44B TAD scenario with EPRI Case 1, worker dose would increase by by 2,028 person-rem due to increased at-reactor package loading; by 1,174 person-rem due to transportation of additional casks; by 701 person-rem due to increased CSNF receipt and handling at Yucca Mountain; and by 155 person-rem to to increased drift excavation to emplace additional waste packages. Compared to EPRI Case 1, DOE's proposal to use the 21P/44B TAD canister for transport, aging and disposal could result in a 4,058 person-rem increase in worker dose.

As shown in Table 3-1, comparing DOE's proposed 21P/44B TAD scenario with EPRI Case 2, worker dose would increase by 2,813 person-rem due to increased at-reactor package loading; by 1,783 person-rem due to transportation of additional casks; by 1,791 person-rem due to increased CSNF receipt and handling at Yucca Mountain; and by 166 person-rem to to increased drift excavation to emplace additional waste packages. Compared to EPRI Case 2, DOE's proposal to use the 21P/44B TAD canister for transport, aging and disposal could result in a 6,553 person-rem increase in worker dose.

4

DOE ASSUMES TOO FEW NON-TAD SHIPMENTS TO YUCCA MOUNTAIN

4.1 Technical Bases

The YMSEIS (DOE, 2008d) assumes that a total of 307 DPCs and storage-only canister-based systems would be shipped to the repository and unloaded at the repository under the 70,000 MTU repository case. In the case that assumes all CSNF is accepted at the repository (referred to in the YMSEIS as Module 1), a total of 966 DPCs are assumed to be shipped to the repository and unloaded at the repository. (DOE, 2008d, Section A.2, Table A-3)

A discussed in more detail in Section A.2, EPRI estimates that utilities could load as many as 2,155 DPCs at reactor sites through 2020. Utilities have also loaded 220 canister-based storage-only dry storage systems – the YMSEIS assumes that some of these canisters would be transported to the repository for repackaging at the repository. Thus, EPRI estimates that as many as 2,375 DPCs and canister-based systems could be storing CSNF by 2020.

4.2 Potential Impacts Associated with Unloading Dual-Purpose Metal Casks and Storage-Only Casks

As discussed in more detail in Appendix A, the YMSEIS does not assume that CSNF stored in dual-purpose metal casks or storage-only metal casks will be transported to the repository and repackaged at repository surface facilities. Therefore, EPRI estimated a worker dose of 35 person-rem associated with unloading dual-purpose metal casks and 26 person-rem associated with unloading storage-only metal casks at reactor sites for repackaging prior to transport to the repository. As noted above, the YMSEIS assumed that 307 to 966 DPCs and/or storage-only canister systems will be transported to the repository for repackaging under the 70,000 MTU repository scenario and the full MTU (DOE 2008d, Module 1) scenario, respectively.

4.3 Potential Impacts due to DOE Assumption of too Few Non-TADs

EPRI estimates that as many as 2,375 DPCs and storage-only canisters could be in use at reactor sites by 2020. If these systems had to be unloaded at reactor sites for repackaging prior to transport, EPRI estimates a unit worker dose of 260 person-mrem per package unloaded, which results in worker doses of 57 person-rem and 560 person-rem for with unloading storage-only canister systems and DPCs, respectively. Thus, if as many as 2,155 DPCs were unloaded at reactor sites, worker dose would increase by 796 person-rem relative to DOE's baseline scenario (307 DPCs; Table A-3) and by 309 person-rem compared to DOE's high-DPCs scenario (966 DPCs; Table A-3). Appendix B.1.4. provides more detail on this estimate.

Occupational Health Impacts at the Reactor Sites

Radiological Impacts:

Table 4-1 summarizes the radiological impacts associated with unloading of various canister systems at the reactor sites.

Table 4 - 1
Radiological Impacts Associated with Unloading of Various Canister Systems at Reactor Sites

Canister System	Worker Dose (person-rem)
307 DPCs/storage-only canisters	80
966 DPCs/storage-only canisters	251
2,375 DPCs/storage-only canisters	560
135 dual-purpose metal casks	35
101 storage-only metal casks	26

Occupational Health Impacts at Yucca Mountain

Radiological Impacts:

- Increased dose associated with unloading DPCs at Yucca Mountain: 135 person-mrem per additional DPC unloaded
- 966 DPCs unloaded compared to 307 DPCs/storage-only canisters assumed in YMSEIS: 89 person-rem
- 2,375 DPCs and storage only canisters unloaded compared to 307 DPCs/storage-only canisters assumed in YMSEIS: 280 person-rem

5
DOE OVERESTIMATED THE PROBABILITY OF IGNEOUS ACTIVITY

5.1 Technical Bases

The geological setting surrounding Yucca Mountain contains several extinct volcanic centers formed over the last 12 million years. DOE has conducted numerous surface and sub-surface investigations of exposed and buried volcanic features to develop a basis for judging the probability of a future volcanic (igneous) event intersecting the proposed Yucca Mountain repository. The results of these investigations have enabled DOE to conduct Probabilistic Volcanic Hazard Analyses (PVHA) to determine if the geological evidence supports a probability of future occurrence below or above the regulatory threshold for consideration of future scenario-initiating events, which is a future occurrence rate of 1 part in 10,000 for a 10,000 year period, or 10^{-8} per year (NRC, 2005). The License Application (DOE, 2008b) uses the probability value obtained in the 1996 PVHA Panel study of 1.7 x 10^{-8} per year (CRWMS M&O, 1996, pp. 4-1), which means this scenario of future volcanism narrowly exceeds the threshold for exclusion in licensing review.

EPRI has recently conducted (EPRI, 2008b, in preparation) an independent assessment of the likelihood of a future volcanic event occurring at the proposed Yucca Mountain repository site. The assessment methodology adopted in the EPRI study was based on same methodology applied in the 1996 Probabilistic Volcanic Hazard Analysis (PVHA) report (CRWMS M&O, 1996, pp. 2-19) and utilized in the LA as noted above. The purpose of EPRI's study was to independently develop new insights and probability estimates for future volcanism based on the more recent, extensive geological and structural data obtained during the last 12 years in the Yucca Mountain region (YMR), especially including recent determination of relatively ancient age (8-10 million years before present) for several buried anomalies in the Yucca Mountain region, which were undated and speculated to be of much younger age in the 1996 PVHA study.

EPRI's PVHA study includes consideration of new geochemical, geophysical, seismological, geodetic and age-dating data collected since the 1996 PVHA report (e.g., Brocher et al., 1998; Day et al., 1998; Perry et al. 1998; Fridrich, 1999; Fridrich et al. 1999; Potter et al., 2002; 2004; Perry et al., 2005; Valentine et al., 2005; 2006; Parson et al., 2006; Valentine and Krough, 2006; Valentine and Perry, 2006; Gaffney et al., 2007; Perry, 2007; Valentine and Perry, 2007; Valentine et al. 2007; Keating et al, 2008), in particular information from the drilling and characterization of various anomalous features buried under alluvial deposits that have been speculated from aeromagnetic data to be additional volcanic centers. Furthermore, EPRI's independent update to the 1996 PVHA report includes consideration of structural factors that demonstrably have controlled the actual eruptive location of volcanic centers that have occurred in the Yucca Mountain region in the last 12 million years (Valentine and Perry, 2006; 2007; Gaffney et al., 2007; Keating et al, 2007). As noted by the NRC's Advisory Committee and Nuclear Waste (ACNW) report on volcanism (ACNW, 2007, pp. 63), for example, there has been no igneous intrusion into Yucca Mountain block in the last 10 million years.

The approach taken by EPRI (EPRI, 2008b, in preparation) follows that used in the 1996 PVHA (CRWMS M&O, 1996). The approach involves defining an igneous event that may intersect the footprint of the proposed repository within the next 10,000 to 1,000,000 years. The calculation requires that an igneous event be well defined and its characteristic features be quantified, and the identification of factors that govern the location and timing of a possible future igneous event in the YMR. By following a similar approach as the 1996 PVHA calculation, results from EPRI's calculation may be compared and evaluated to results in the 1996 PVHA (CRWMS M&O, 1996) and a planned PVHA-U (the updated version of the 1996 PVHA) by the USDOE. Appendix F provides a more detailed discussion of the methodology EPRI used in its PHVA.

EPRI's independent PVHA work finds the 1.7 x 10^{-8} per year probability of a future igneous event intersecting the proposed Yucca Mountain repository used in DOE's TSPA License Application (OCRWM, 2008) to be an overestimate. A more reasonably expected value of 3.0 x 10^{-9} per year, with a range of 0.0 to 7.3 x 10^{-9} per year for the period between 10,000 and 1,000,000 following repository closure, is supported by recent independent analyses based on up-to-date, site-specific information and models (EPRI, 2008b, in preparation). The implication of this lower probability value is that consideration of future igneous/ volcanic events occurring at Yucca Mountain fall below the regulatory threshold for inclusion in licensing review.

5.2 Potential Impacts due to Overestimating the Probability of Igneous Activity

The draft EPA and NRC regulations for Yucca Mountain specify that if the probability of a particular event, such as igneous activity within the Yucca Mountain repository footprint, is less than one chance in 10,000 over 10,000 years, then the consequences of such an event need not be evaluated (EPA, 2005; NRC, 2005). DOE's overestimation of the probability of igneous activity at Yucca Mountain could lead to an outcome EPA specifically intended to avoid with its "reasonable expectation" approach, i.e., consideration of unlikely events at cost of "deflect[ing] attention from questions critical to developing an adequate understanding of the expected features, events, and processes."

Furthermore, the DOE estimates of igneous consequences in the licensing process may be subject to considerable regulatory scrutiny. The mean dose to the Reasonably Maximally Exposed Individual (RMEI) living downstream of Yucca Mountain due to igneous activity scenarios is the dominant contributor to overall dose to the RMEI from all scenarios[DOE LA, 2008b]. Therefore, NRC and, potentially, third parties to the licensing process may review the igneous consequence analysis work in great detail. This may extend the time to complete the licensing process.

It is difficult to link DOE's overestimation of the probability of igneous activity to specific outcomes of the licensing process that lead directly of negative impacts on worker health and safety. However, it is conceivable that by further complicating an already complex analysis and licensing task with inclusion of igneous activity its License Application, DOE has increased the likelihood that the shipment of CSNF from reactor sites and other commercial facilities will be subject to further delay. Any additional delay adds to the occupational health risk borne by workers at the storage sites.

The need to store additional amounts of CSNF for an additional amount of time will increase both radiological and non-radiological health risk primarily to workers at the reactor sites due to additional CSNF handling and monitoring in both dry and wet storage. Storage of additional

CSNF at reactor sites will also have a radiological impact on members of the public that may live near the at-reactor dry storage location(s).

For each year of delay in the start of acceptance of CSNF by DOE, nuclear utilities will have to load additional CSNF into dry storage canisters – most likely TAD canisters. Solely for the purposes of estimating occupational health risk consequences, EPRI assumes that once DOE begins repository operations, DOE would provide nuclear utilities with TAD canisters and transportation casks for shipment of CSNF offsite.

The NWPA limits Yucca Mountain capacity to 70,000 MTHM of CSNF and DOE spent nuclear fuel and HLW, 63,000 MTHM of which is available for disposal of CSNF. The nuclear utilities will soon exceed this waste inventory. Accordingly, CSNF that is discharged from reactors above and beyond the 63,000 MTHM limit does not have a final disposal pathway even with an operational Yucca Mountain unless the legislatively mandated disposal capacity is increased or until another repository becomes available.

Appendices B and C of this report provides an assessment of the potential radiological and non-radiological occupational health impacts of a one-year delay in the initiation of CSNF shipments to Yucca Mountain. Table 5-1 provides a summary of key radiological and non-radiological impacts resulting from a one-year delay in the availability of Yucca Mountain to begin receiving CSNF from reactor sites industry-wide. In addition, if existing ISFSI storage space is consumed or ISFSI storage does not exist, there would be additional occupational risk associated with the construction of a new ISFSI storage pad.

Table 5 - 1
Summary of Industry-Wide Occupational Impacts Due to a One-Year Delay in the Availability of Yucca Mountain (Based on 75 Reactor Sites)

ISFSI Activity	Dose (person-rem)	Injuries and Fatalities (cases)
Surveillance and inspection	9	0.052 TRC 0.027 LWC 4.1×10^{-5} fatalities
Maintenance	112.5	0.052 TRC 0.027 LWC 4.1×10^{-5} fatalities
Additional storage module construction at existing ISFSI	27 – 37	7.5 – 10 TRC 4.2 – 5.7 LWC 0.013 – 0.0189 fatalities

Radiological impacts arise to routine ISFSI operations, totaling approximately 120 person-rem with incremental increases in risk due to non-radiological hazards faced by a utility worker. The construction of additional dry storage modules, as illustrated in Table 5-1 and described in more detail in Appendices B and C, also result in significant increases in worker risk associated with ISFSI expansion.

In the event that either existing ISFSI pad capacity at a particular site is full or does not exist, the construction of a new pad could become necessary. The occupational consequences associated with the construction of one ISFSI pad at a reactor site (from Section C.1.3) is estimated as:

- 22 TRC
- 12 LWC
- 3.9×10^{-4} fatalities

Economic Impacts

In addition to occupational impacts, the further delays of CSNF shipments to Yucca Mountain could also potentially lead to significant costs to the utilities. EPRI expects that between 80% and 100% of CSNF discharged after 2020 will require an equivalent amount of CSNF to be loaded into dry storage. If DOE does not begin repository operations and the subsequent acceptance of CSNF by that time, EPRI assumes that nuclear utilities will have to procure TAD canisters for this additional CSNF that requires on-site storage. Thus, any additional delay in the start of repository operations will result in an economic impact for the nuclear utilities to cover the additional cost of CSNF handling and monitoring, as well as the economic impact associated with the purchase of additional TAD canisters for on-site storage. Appendix *G* provides an assessment of the potential economic impacts of a one-year delay in the initiation of CSNF shipments to Yucca Mountain. These impacts are summarized below:

- Incremental cost of additional TADs to the utilities: $0.75 million per canister, plus $300,000 per storage overpack;
- Cost of additional TAD transfer and monitoring operations at reactor sites: $150,000 to $300,000 per TAD loaded.

Table 5-2 summarizes potential occupational and economic impacts due to a one-year delay in CSNF shipments to Yucca Mountain.

Table 5 - 2
Summary Occupational and Economic Impacts of a One-Year Delay in the Availability of Yucca Mountain

Health or Economic Risk Category	**Health Risk Type**	**Metric of Worker Health or Economic Impact**	**Lower value**	**Upper value**
Reactor workers	Radiological	[person-rem]	149	159
	Non-radiological	(cases)		
		▪ TRC	30	32
		▪ LWC	16	28
		▪ fatalities	0.013	0.019
Economic [$]	Cost of additional TAD canisters and storage overpacks at reactor sites	Unit Cost per TAD and Overpackg (Millions $)	$1.05	$1.05
	Cost of loading additional TAD canisters at reactor sites	Unit cost per TAD loaded (Millions $)	$0.15	$0.30

6
DRIP SHIELDS ARE NOT NEEDED

6.1 Technical Bases

There are several conservatisms in DOE's analyses of post-closure performance that have led DOE to unnecessarily include drip shields in its repository design. These conservatisms include:

1. Overestimation of the amount of net infiltration, thereby incorrectly indicating a larger benefit of the use of a drip shield than is actually the case;
2. Overestimation of the fraction of the repository experiencing seepage into the open drifts, having the same effect as overestimation of net infiltration;
3. Overestimation of seismic energy and rockfall. This leads DOE to the conclusion that drip shields would provide significant protection from rockfall;
4. Overestimation of damage to the TADs due to seismic and rockfall events. This also leads to the incorrect conclusion that drip shields would be required to provide additional protection from damage of the waste packages;
5. Overestimation of the rate at which Alloy 22 (part of the waste package (WP)) will degrade. This, in turn, gives greater performance credit to the drip shields than is warranted.
6. Cladding performance has been neglected. EPRI analyses indicate that including credit for the performance of the CSNF cladding in the dose analysis is appropriate and that such inclusion would provide an additional barrier to the release of radionuclides from the waste form. This, in turn, would also reduce the need for a drip shield;
7. Performance of the stainless steel barriers (i.e., the inner WP cylinder and the outer shell of the TAD) in the waste package has been neglected. Including performance of these components in the overall performance analysis would also reduce the need for a drip shield.
8. DOE notes that it typically uses the more conservative of two or more conceptual models. Some of these conservatisms could also result in the apparent need for drip shields. As a consequence of this general approach, each conservatism is compounded by conservatisms in other parts of the analysis. Therefore, each of the conservatisms identified here, significant in their own right, compound each other to produce a very large degree of conservatism.

Each of these issues will be discussed in the following subsections

6.1.1 DOE Overestimated Net Infiltration

Both DOE and EPRI have taken the position that there will be three climate states during the next 10,000 years. The definitions of these states are either the same or somewhat similar:

- DOE's "Present-day" and EPRI's "Interglacial" climate states are essentially the same. DOE assumes the "present-day" climate will exist from the time of repository closure to 600 years after closure; EPRI assumes its "interglacial" state will occur from 1000 to 2000 years after repository closure.
- DOE's "Monsoon" climate and EPRI's "Greenhouse" climate states are roughly the same in that both of these climate states assume warmer and wetter conditions in the Yucca Mountain region. DOE assumes the "monsoon" climate will exist from 600 to 2000 years after repository closure; EPRI assumes its "greenhouse" state will occur from the time of repository closure to 1000 years after closure.
- DOE's "Glacial transition" and EPRI's "Full Glacial Maximum" (FGM), while both representative of a cooler, wetter climate than exists today in the Yucca Mountain region, are not exactly the same. While DOE notes that the past coldest glacial states are OIS 16, 12, 6, and 2, which could provide the largest amount of net infiltration and seepage, DOE defines its "glacial-transition" climate to be the transition between OIS 11 and OIS 10. (DOE, 2008b, Section 2.1.2.1.1). As these two climate states are similar, it could be expected that EPRI's choice of the FGM would result in higher amounts of net infiltration and seepage than DOE's "glacial-transition" climate state. Both DOE and EPRI assume the "glacial-transition"/FGM state will occur from 2000 to 10,000 years after repository closure.

A comparison of net infiltration values used by DOE and EPRI is presented in Table 6-1. Since the publication of EPRI's IMARC-8 report (EPRI, 2005a), EPRI numbers in bold italic type have been adopted in its TSPA for all times as sensitivity studies indicate no sensitivity to net infiltration rates during the first 2000 years for the Base Case (no seismic, rockfall, or igneous events), and little sensitivity during the first 2000 years for the Base + Seismic/Rockfall and Base + Igneous Intrusion Cases.

EPRI's best estimate values for net infiltration (EPRI, 2005) are lower than the values used in DOE's license application for all climate states (DOE, 2008b). Hence, EPRI believes that DOE has overestimated net infiltration averaged over the Yucca Mountain repository footprint.

One of the main arguments for the use of drip shields is to reduce the amount of groundwater entering the disposal drifts. As DOE has overestimated net infiltration, this results in an overstatement of the positive effect of the drip shields with respect to long-term repository performance.

Table 6 - 1
Comparison of DOE and EPRI Net Infiltration Rates (mm/y) [Sources: DOE (2008b), Tables 2.3.1-2 through 2.3.1-4 "Repository footprint" values; EPRI, 2005a)]

Climate State	**Time Period [years after closure]**		**Mean-1 s.d./Min (DOE) or Low (EPRI, P=0.05) Value**		**Mean (DOE) or Moderate (P=0.9) / Probability-weighted (EPRI) Value**		**Mean+1 s.d./Max (DOE) or High (EPRI, P=0.05) Value**	
DOE and EPRI Climate State Name	DOE	EPRI	DOE (Mean - 1 s.d./Min)	EPRI (Low)	DOE Mean	EPRI (Moderate / Prob.-weighted)	DOE (Mean + 1 s.d./Max)	EPRI (High)
"Present Day" (DOE); "Interglacial" (EPRI)	0-600	1000-2000	5.1/1.5	1.1	17.6	7.2/7.0	30.1/48.2	9.6
"Monsoon" (DOE); "Greenhouse" (EPRI)	600-2000	0-1000	9.6/1.2	1.1	32.9	11/11	56.2/95.3	19
"Glacial-Transition" (DOE); "Full Glacial Maximum" (EPRI)	2000-10^6	2000-10^6	17.4/4	***6.8***	38.6	***20***/20	59.8/97.3	***35***

Notes: A direct comparison of values is not possible as EPRI uses a logic tree approach whereas DOE uses a continuous distribution. EPRI assigns a probability of 0.05, 0.9, and 0.05 for the Low, Moderate, and High infiltration rate values, respectively. Hence, the closest comparison would be between DOE's Mean and EPRI's Probability-weighted values. However, the table also compares DOE's "Mean minus 1 standard deviation (s.d.)" and "Minimum" values ("Mean – 1 s.d./Min") to EPRI's "Low" value, and compares DOE's "Mean plus 1 s.d." and "Maximum" values ("Mean + 1 s.d./Max") to EPRI's "High" value.

6.1.2 DOE Overestimated Seepage Rates

Table 6-2 provides a general comparison of the seepage fractions and seepage rates (averaged over all waste packages) for intact drifts (no rockfall) for the three climate states that are postulated by DOE and EPRI. Although difficult to compare directly due to the probabilistic complexity of the DOE seepage model (see the second and third notes under the table for the comparisons EPRI used), EPRI has determined that DOE has significantly overestimated the amount of seepage that would occur into the disposal drifts. Thus, EPRI concludes that DOE's seepage fraction and seepage rate estimates are conservative. Overestimates of seepage fractions and rates will also overstate the potential benefit of using drip shields as one of the purposes of the drip shields is to reduce WP seepage rates.

Table 6 - 2

Comparison of DOE and EPRI Seepage Fractions and Seepage Rates (Maximum Likelihood Flow Field (DOE) Seepage Case (EPRI); Mean (DOE) or Probability-weighted (EPRI) Net Infiltration). [Sources: DOE (2008b); EPRI, 2005a)]

Climate State [DOE/EPRI]	Seepage Fraction (%)		Seepage Rate (kg/yr/WP)*	
	DOE**	EPRI Probability-weighted Seepage Case***	DOE Mean**	EPRI Probability-weighted Seepage Case***
Present-day/Interglacial	1.1	0.33	1.2	0.50
Monsoon/Greenhouse	2.2	0.33	4.6	0.93
Glacial Transition/Full Glacial Maximum (FGM)	4.7	0.44	14.4	1.9

Notes:
*Averaged over all waste packages.
**10th percentile infiltration scenario (maximum likelihood scenario), Section 2.1.2.1.2, (DOE, 2008b)
***Probability-weighed seepage fraction/rate: Base Seepage Case (P=0.96): High Seepage Case (P=0.04)

6.1.3 DOE Overestimated the Amount of Seismic Energy and Rockfall

DOE also indicates that the presence of drip shields will protect the underlying waste packages in the event of rockfall due to thermal stresses or seismic events. The higher the estimate of rockfall, the more beneficial it would seem to install drip shields.

However, EPRI has determined that DOE overestimated the amount of rockfall that will occur for these two mechanisms during the first several hundred thousand years following repository closure (EPRI, 2005b; 2006b). EPRI determined the extent of rockfall (dynamic and static) versus time by dividing the repository into eight rock property categories. In addition to dynamic rockfall during seismic events, long-term stress corrosion cracking of the rock was also considered. Combining the effects of dynamic and static rockfall, along with waste package (WP)-to-WP collisions, over a series of ten seismic events results in only a modest increase in the number of WP failures that occur compared to the nominal scenario (no disruptive events). Thus, adding the multiple seismic event scenario to the nominal scenario increases the probability-weighted peak individual dose by less than a factor of two (EPRI, 2005b). The results from these EPRI analyses are:

- Dynamic rockfall produces inconsequential effects on the waste packages, even for large rock sizes,
- Static effects of rocks on the waste package are inconsequential for credible stresses and maximum extent of potential drift collapse, and
- WP-WP collisions produce damage to the internal lid from impacts with the waste package internals. The outer lid, however, was undamaged by the collisions."

6.1.4 DOE Overestimated the Amount of Damage to TADs due to

Seismic/Rockfall Events

An important clarification regarding the seismic ground motion modeling case is that the releases and annual doses for the 10,000-year time period are only for the damaged co-disposal waste packages. As described in Section 2.4.2.2.2.3 of DOE (2008b), the releases from the commercial SNF waste packages contribute only negligibly to the total dose of the seismic ground motion modeling case because of the low consequences of seismic-induced failures of commercial SNF waste packages. Seismic-induced failures of commercial SNF waste packages result in low consequences largely due to the low probability of damage to TADs bearing commercial SNF in the first 10,000 years. The expected damage frequency for TADs bearing commercial SNF is calculated to be 5.249 × 10-9 per year, which leads to the probability of failure of 5.249 × 10-5 in 10,000 years (DOE, 2008b, pg. 2.4-57). Thus, DOE determines the probability-weighted number of SNF WPs that would fail due to seismic damage during the first 10,000 years is less than one.

The occurrence of seismic events is described as a Poisson process with the highest annual exceedance frequency, max, of potentially damaging events equal to 4.287 × 10-4 per year and the lowest annual exceedance frequency of min equal to 10-8 per year (DOE, 2008a), which is the threshold in proposed 10 CFR 63.342(b) for the occurrence rate of very unlikely events that can be excluded from the performance assessment. Based on these exceedance frequencies from the seismic hazard curve, the expected number of events in any time period T is equal to (max • min)T. Thus, during the first 10,000 years after permanent closure, approximately four potentially damaging events can be expected to occur, compared to approximately 430 potentially damaging events in the 1,000,000-year period after permanent closure (DOE, 2008a).

6.1.4.1 DOE Overestimated Seismic Energy

DOE uses ground motions estimated from its Yucca Mountain Probabilistic Seismic Hazard (PSH) model (Stepp et al. 2001). The seismic hazard curve in Stepp et al. (2001) is reproduced here as Figure 6-1. At return periods of 10^6 years, the Yucca Mountain PSH model predicts a mean PGA and PGV of 3g and 400cm/sec, respectively. These are ground motions that exceed the largest magnitudes ever recorded in the world, so there is some uncertainty as to whether they are physically realistic (Bommer et al. 2004). The PGA curve (presented in Figure 1.7-7 of DOE, 2008b) and reproduced here as Figure 6-1, is an extrapolation of the PSHA curve to 10^{-6}/year and beyond. It is important to recognize that a statistical distribution is just a model of observed data, and extrapolation beyond the range of the data may not be valid. EPRI asserts that the extrapolation of the maximum horizontal acceleration is beyond the region that could be supported by the strength of the rock and soil at Yucca Mountain. In a review of the results of this PSHA, an expert panel convened by the USGS (Hanks, et al., 2006), concluded the following:

> As an overall and quite general finding – and also as a brief summary of the findings that follow – the Committee finds that there are many lines of evidence and argument that can be drawn from a wide range of geological, geophysical, seismological, and material-properties studies that all point to the same general conclusion: at probabilities of exceedance of 10^{-4}/yr and smaller, the seismic hazard at Yucca Mountain as calculated from the 1998 PSHA is too high.

Similarly, a limitation found in the analyses of earthquake ground motion input for Yucca Mountain preclosure surface seismic design and post closure performance (MDL-MGR-GS-000003 Rev 01) states:

> While these ground motions can be used to assess the sensitivity of the response of waste emplacement drifts and engineered barrier system components to such high levels of motions, ultimately results should be evaluated for ground motions that are credible for Yucca Mountain.

This statement reflects the fact that even the authors of the ground motion assessment at Yucca Mountain believe that their results are too high and not credible for design. Their use is only recommended for sensitivity studies. Therefore, it is reasonable to conclude that the consensus in the community of earthquake professionals that ground motion estimates at Yucca Mountain are too high at probabilities of 10^{-6} /year and should be lower.

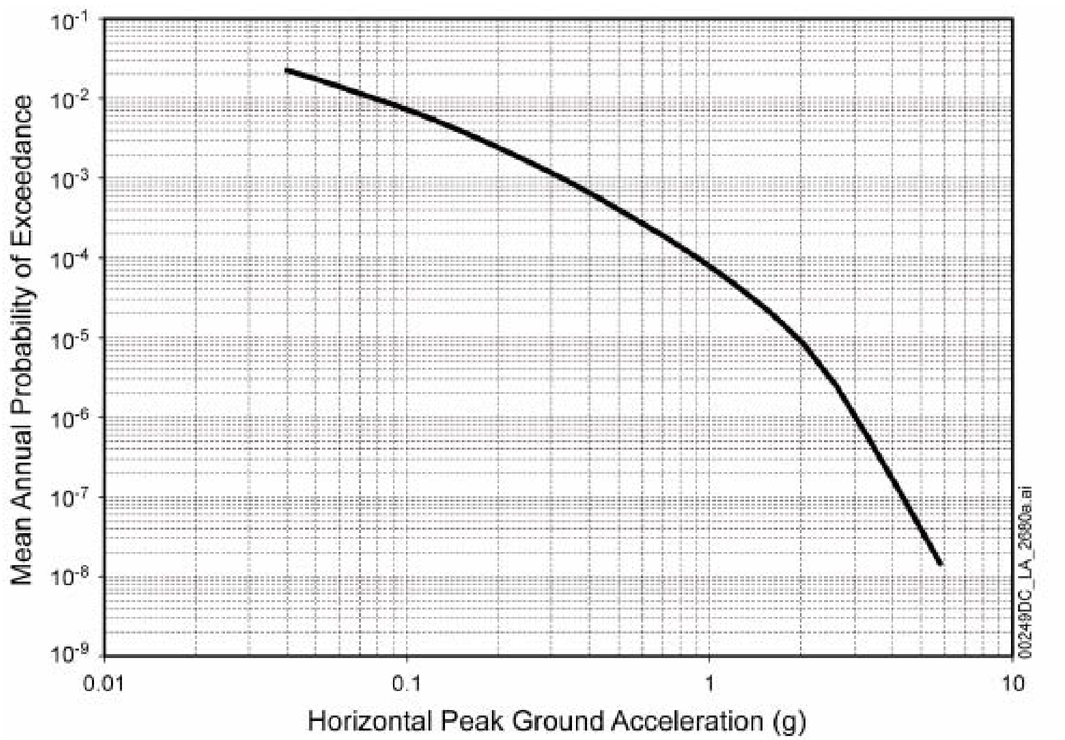
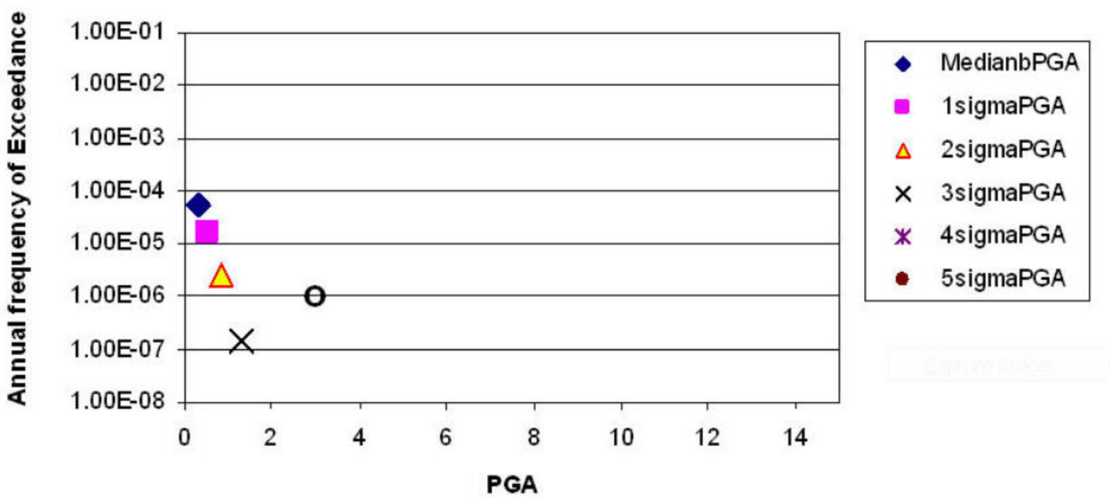


Figure 1.7-7. Seismic Hazard Curve Used in the Preclosure Safety Analysis for Surface Facilities

Figure 6 - 1
DOE Seismic Hazard Curve Adapted for Post-closure Use [reproduced from Stepp et al. (2001), Figure 1.7-7]

Logically, the closest, most active earthquake sources to Yucca Mountain should be responsible for the largest ground motion levels, and EPRI's analysis compared the ground motion levels of these sources to those of the Yucca Mountain PSH model (EPRI, 2006b). Therefore, EPRI considers the Solitario Canyon Fault (SCF) to be the most important fault upon which to base future seismic activity estimates. EPRI also considers one "background fault" in its analyses (EPRI, 2006b).

Figure 6-2 shows EPRI's estimates of the annual frequency of exceedance for PGA and PGV for the SCF and a background earthquake. Each horizontal line of three matching symbols on Figure 6-2 reflects the range of magnitudes estimates for the SCF (EPRI 2006b, Table 2-1). The open circles on the graphs represent the mean PGA and PGV for the 10^6 year return period from the Yucca Mountain PSHA (Stepp et al., 2001). The analysis shows the PGA to be about 0.7 to 1g for the SCF at the 10^{-6}/yr annual frequency of exceedance, (10^6 year return period), considerably less than the 3g estimated from the Yucca Mountain PSH model for the same return period. A PGV of 70 to 160 cm/sec is estimated for the SCF at the same annual frequency of exceedance or return period, considerably less than the 400 cm/sec derived from the Yucca Mountain PSHA. Similar results are obtained for the background earthquake (Figure 6-2).



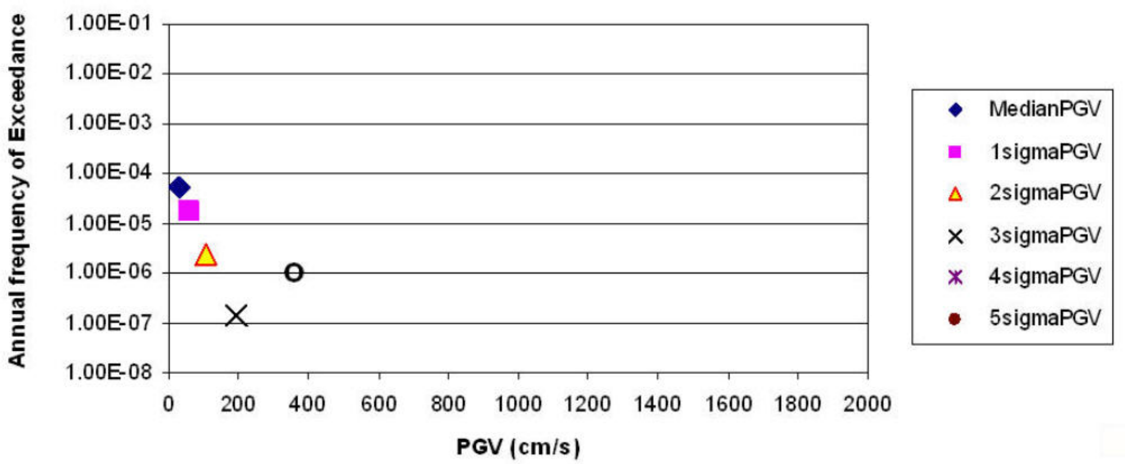


Figure 6 - 2
EPRI's hazard estimates for the Solitario Canyon Fault (upper figure) and background earthquake (lower figure) sources. The open circles show for comparison the mean values for the 10-6/year annual frequency of exceedance (106 year return period) from the Yucca Mountain probabilistic seismic hazard model (Stepp et al. 2001).

Therefore, EPRI has chosen to apply a 0.75 m/s peak ground velocity (PGV) with a 10^5 year recurrence interval, so that repeated seismic events have been stylized as 10 large events over a 10^6 year period, spaced out equally in time (EPRI, 2006b). These large events are those that have been judged most likely to produce changes in the repository that may alter its long-term performance.

6.1.4.2 DOE Overestimated Waste Package Damage due to Seismicity and Rockfall Events

It is EPRI's position that waste package damage is limited due to seismic and rockfall events for cases involving either the presence or absence of drip shields. EPRI reaches this conclusion even for very large events that occur when the waste package outer barrier is degraded; small events, even if frequent, are expected to produce minimal damage to the waste packages. Smaller events occurring with greater frequency are less likely to be of importance to the Total System Performance Assessment (TSPA).

EPRI (2005b; 2006b) considered the effects on WP integrity for the following cases:

- WP-to-WP collisions due to seismic ground motion with PGVs of either 0.75 m/s or 2 m/s, drip shields in place, either flat-on or oblique WP-to-WP contact;
- Dynamic rockfall directly onto the center of a WP, drip shields absent;
- Static rock rubble loading directly on a WP, drip shields absent, Alloy 22 outer shell either present or absent.

EPRI (2005b) notes however that DOE's own analyses suggest that little rockfall will occur for the first 20,000 years:

> The DOE approach to modeling time-dependent rock degradation in the lithophysal units at Yucca Mountain is judged by EPRI to be reasonable and utilizes the most up-to-date knowledge on time-dependent rock mechanics and numerical techniques. … DOE's results indicate little rockfall is expected out to 20,000 years after waste emplacement due to time-dependent processes alone. Other [DOE]results … also indicate that, when combined with thermal loading and seismicity, time-dependent loss of rock cohesion up to 20,000 years is not a major contributor to rockfall. Note, however, that the DOE approach involves basing the UDEC time-dependent model on an exponential formulation of the stress corrosion law without a lower threshold stress limit and use of material properties for heated rather than ambient temperature tuff. These are clearly conservative assumptions, hence, DOE's results … represent pessimistic upper bounds on possible rockfall for the period of 10,000 to 20,000 years after repository closure.

Therefore the drip shields are not needed to protect the WPs from rockfall for the first 20,000 years following permanent closure or more.

WP-to-WP Collisions

Two sets of impact analyses for adjacent waste packages are discussed in EPRI (2006b): an analysis of a collision into an unyielding surface at 2 m/s and an analysis of a collision into an unyielding surface at 0.75 m/s. Use of an unyielding surface is conservative in that this assumes two adjacent waste packages are traveling in opposite directions, each with a velocity of either 2 or 0.75 m/s.

For 2 m/s PGV, plastic deformation leading to residual stresses does not develop in the WP outer Alloy 22 shell for a flat-on impact between two waste packages (EPRI, 2006b). Some yielding develops on the inner stainless steel lid and around the connection of the inner lid with the inner stainless steel shell, but this would not affect the performance of the waste package. For an oblique impact where a waste package is tilted at 4 degrees such that the impact is along an edge of the outer lids, some yielding develops in the outer lid under the reduced impact area. Yielding with plastic deformation and residual stresses also develops at the connection of the middle lid to the outer Alloy 22 shell. Such yielding leads to a potential for tearing of the weld at the middle lid connection if the waste package experiences impacts at this PGV multiple times over the life of the waste package. An extrapolation of these results would indicate that the potential for tearing the middle lid connection and yielding in the outer lid should be reduced to a very small probability below an impact of about 1 m/s.

For a PGV of 0.75 m/s, some minor plastic deformation develops on the outer shell in a small area under the concentrated load for the oblique (worst-case) impact orientation. However, no residual damage occurs in the inner or middle lids or in the closure connections for these lids. Thus, it can be concluded that even multiple impacts at this 0.75 m/s impact velocity for the worst-case orientation would not lead to eventual tears or failure of the inner lid as a containment boundary. Although some plastic deformation of the outer shell is predicted for a PGV of 0.75 m/s, this deformation results from compressive loading, so neither immediate structural failure nor delayed SCC penetration is expected. In addition, the extent of damage is so small that even repeated impacts are not expected to lead to a breach in containment.

When the WP inner SS and TAD outer SS shells are intact, DOE reaches a similar conclusion: "Note that for the CSNF WP with intact internals [SCC] damage [due to WP-to-WP collisions] occurs only at the 4.07 m/s PGV level (the probability is zero for all other PGVs" (DOE, 2008b, pg. 6.6-13). For more reasonable PGV values (EPRI, 2005b; 2006b), even DOE finds there will be *no* WP-to-WP damage during seismic events. Hence, both DOE and EPRI conclude the presence or absence of drip shields has no effect on WP damage due to WP-to-WP collisions during seismic events.

DOE also considers a scenario in which the outer containment barrier (OCB, the Alloy 22 shell) could be punctured by sharp WP internals caused by degraded internals. While DOE conservatively concludes that OCB punctures are more likely than SCC failures due to the rubble loading, at more reasonable seismic energy values (PGV less than approximately 1 m/s), even DOE shows essentially no WP damage due to either SCC or internal puncture (DOE, 2008b, Figures 6.6-14 and 6.6-17). Thus, DOE's conservative internal puncture analyses would also inappropriately heighten the value of including drip shields in the repository design.

Dynamic Rockfall

For dynamic loading, EPRI (2005b) conservatively assumed that a large rock block is ejected directly onto the top of a bare WP (i.e., without the DS present), as shown in Figure 6-3. EPRI used an Alloy 22 thickness of 20 mm. The rock block EPRI modeled was assumed to be 7.49 metric tons with a volume of 3.11 m^3. This size of rock is the largest size in a representative grouping considered to have a reasonable probability of occurring for the maximum PGV of 2 m/s that EPRI has determined should be associated with a future seismic event near the Yucca Mountain site. This block was assumed to be ejected with a downward velocity of 2 m/s. Furthermore, EPRI (2005b) conservatively assumed that the rock block struck the unprotected WP on a knife edge (see Figure 6-3). EPRI has concluded that even in the event of the postulated occurrence, the WP internals would not be degraded in a manner such that they would fail to provide sufficient structural support to protect the contents. EPRI (2005b) concludes that:

> [A] rockfall impact event with the largest size rock in a representative grouping considered to have a reasonable probability of occurring for the maximum PGV of 2 m/s associated with a future seismic event will have very little effect on the longevity of the Alloy 22 WP outer shell. The response of the Alloy 22 material under the impact will likely remain in the linear regime, even with some corrosive thickness reduction, and thus, residual stresses that could accelerate the degradation from stress corrosion cracking will not be present. It seems especially evident that if residual stresses near the yield strength of the material are needed for stress corrosion cracking, then such a rockfall event will most certainly not affect the performance or longevity of the waste package.

Hence, it is EPRI's position that dynamic rockfall directly onto a WP – without the presence of a DS – will not cause any additional damage compared to the case for which a DS was present. Thus, drips shields are not needed to protect a WP from dynamic rockfall.

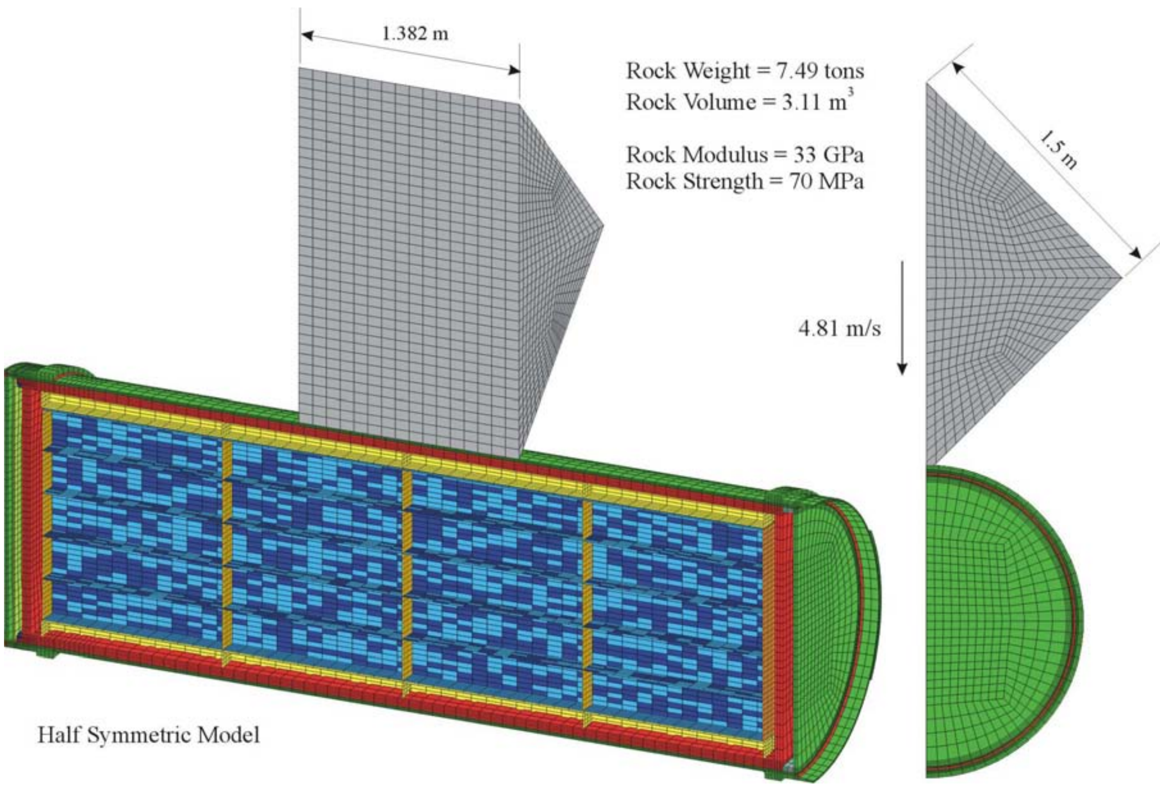


Figure 6 - 3
Finite Element Model and Analysis Setup for Impact due to Rockfall (taken from Figure 12-1 in EPRI (2005))

Analyses were performed to assess the effect of multiple seismic events on the integrity of the engineered barrier system (EBS) (EPRI, 2006b). The analyses are intended to approach a reasonable expectation case, although it is acknowledged that a number of conservatisms remain in the analysis, as a PGV of 0.75 m/s would not be expected to displace the DS. Hence, this analysis does not include the presence of drip shields. Furthermore, it is very conservatively assumed that each large block described above that is ejected leads to the dynamic structural failure of a single WP. The impact of this conservatism is increased when it is assumed that no drip shields are emplaced.

Static Rock Load

EPRI (2005b) estimates that the maximum bulking height for rubble would be in the range of 5 to 20 meters. EPRI uses this amount of bulking to assess the static load and structural response of the WP.

For EPRI's static rubble analysis (EPRI, 2005b), EPRI considers the structural response of degraded waste packages due to static loads from rubble that would pile up on top of the waste package from a chimney-type collapse of a portion of the emplacement drift. No credit is taken for the drip shield and the Alloy 22 waste package outer barrier (WPOB, the outer Alloy 22 shell). Only the bare stainless steel WP inner shell is considered to be in place as the last structural barrier for protecting the spent fuel.[1] This bounding assumption was made to evaluate whether the structural strength of the inner 316 SS WP shell is sufficient to withstand the maximum credible load of rock resting on the WP. If the rubble static load can be withstood by just the SS inner shell, then it could be concluded that the rubble will not cause early WP failure due to structural failure.

EPRI (2005b) concludes that:

> [A] "bare" WP inner shell can survive the static loads that could develop from a collapse of the emplacement drift at Yucca Mountain for a conservative minimum of a 30-m-high pile of rock rubble. As the bare stainless steel inner shell will remain linear for 30m of rubble, it is extrapolated that a waste package with all or part of the Alloy 22 outer shell present (pristine or partially degraded) will also remain linear for a static load of at least 30m of rubble. The loading from a 30m column of rock conservatively calculated as necessary to mechanically fail a degraded WP far exceeds the loading from a 5-20m column of rock that can possibly be developed in degraded drifts at the Yucca Mountain repository due to rockfall and bulking.

Thus, EPRI's position is that drip shields are also not needed to protect the WP from early structural failure due to the maximum expected rubble height.

This conclusion is echoed by DOE:

> The probability of rupture [structural failure] for the 23-mm-thick OCB with intact internals was determined to be zero. … Damage for WPs with intact internals was not calculated for WPs surrounded by rubble. A WP becomes surrounded by rubble after DS framework and DS plates have failed during a seismic event. This is expected to occur at late times after repository closure. … Therefore, CSNF WPs are not likely to have degraded internals at the time of DS failure. (DOE, 2008b, pg. 6.6-14)

However, given that DOE assumes DS failure occurs fairly late in the period of regulatory interest such that some WP corrosion failure may have already occurred, DOE conservatively assumes that groundwater has previously penetrated the WP and degraded the WP internals to

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[1] This compares to DOE's estimate of the minimum WPOB thickness to be considered for rubble load analyses: "[T]his estimate indicates that the 17-mm-thick OCB provides a reasonable representation for seismic response at the end of the period for assessment of repository performance." (DOE, 2008b, pg. 6.6-12)

the point at which DOE assumes the internals provide no structural support (DOE, 2008b, pg. 6.6-14). Thus, it is not possible to compare EPRI and DOE WP structural failure rates due to the presence of rubble as DOE has conservatively assumed the SS internals to the WP provide no structural support.

Cracking of the WP outer barrier due to the static load of the maximum rubble height could occur if the necessary prerequisites for SCC are met; namely: a tensile stress greater than the threshold stress for SCC, a suitable aqueous environment, and a corrosion potential (E_{CORR}) greater than the threshold value for cracking. In EPRI (2006b), only a fraction of the WPs subject to static loading are consider to fail by SCC. First, only those WPs subjected to a static load from a rock pile >10 m in height are considered to sustain a tensile load greater than the threshold for SCC. This height is a conservative estimate based on the height of the rock pile necessary to induce plastic strain for <u>an unprotected inner stainless steel vessel</u> (40 m for uniform loading of the vessel over a 120^o arc), taking into account the stress concentration resulting from point or line loading. This latter effect is simulated using a "stress-concentration factor" of four, based on analyses performed by DOE (BSC, 2004). This estimate conservatively ignores the strength of the Alloy 22 outer barrier itself in determining the necessary height of the rock pile.[2] Second, of those WPs covered by a rock pile >10 m in height, only 71% are assumed to be exposed to an appropriate aqueous environment. Third, only a fraction of the WPs that meet both the threshold stress and environment prerequisites will also exhibit a sufficiently positive E_{CORR} for SCC. EPRI (2006b) concludes that the overall fraction of WP subject to a rock pile >10 m in height that are susceptible to SCC is, therefore, 0.017 (71% of environments multiplied by the 0.024 probability that E_{CORR} exceeds the threshold potential for SCC).

Conclusion of EPRI Seismic and Rockfall Analyses

A total of 64 waste packages are predicted to fail as a result of the repeated seismic events, 18 as a result of dynamic rock impacts and 46, out of a total of 2734 that will be covered by a rock pile greater than ten meters in height, as a result of seismic-induced SCC of the outer barrier (EPRI, 2006b). All of these failures are predicted to occur during the first seven seismic events, with no further drift degradation predicted after 650,000 yrs. The number of dynamic failures decreases with time as the number of large ejected blocks diminishes with each subsequent event. In contrast, the number of static load failures tends to increase with time as more of the drift collapses.

In conclusion, it is EPRI's opinion that a series of conservatisms in DOE's seismic hazard and subsequent rockfall and WP damage analyses has led DOE to believe that drip shields offer some protection to the underlying WPs such that WP failure rates are reduced. EPRI analyses (EPRI, 2005b; 2006b) performed for more reasonable seismic energies and rockfall dynamic and static loads, although still maintaining some conservatism, conclude that excluding drip shields from the repository would have no effect on WP longevity.

6.1.4.3 DOE Finds Drip Shields can Cause WP-to-WP Collision Damage

According to DOE, the presence of drip shields also has the effect of potentially increasing the amount of WP damage due to seismic events. DOE analyses indicate that if the drip shield is present during a significant seismic event, then some WPs will be damaged due to WP-to-WP

[2] Corrosion resistance of the Alloy 22 is not ignored, however.

collisions (DOE, 2008b) Section 6.6 of DOE(2008a) discusses DOE's approach to estimating DS and WP damage due to seismic events. The seismic events considered in the TSPA-LA (DOE, 2008a) are

> Dynamic loads on WPs free to move during a seismic event have the potential to result in a rupture (tear) of a WP if the local strain exceeds the ultimate tensile strain. Dynamic loading from a single impact may not produce tensile strains in the Alloy 22 outer corrosion barrier (OCB) that exceed the ultimate tensile strain. However, the extreme deformation from a major seismic event could weaken the OCB, potentially resulting in a ruptured OCB from a subsequent extreme seismic event. …
>
> The probability of rupture for WPs with degraded internals surrounded by rubble is zero because the strain on the OCB is always below the ultimate tensile strain for Alloy 22. … However, a severely deformed OCB may be punctured by the sharp edges of fractured or partly degraded internal components. The WP internals are assumed to degrade as structural elements after the OCB is first breached.

In contrast, EPRI (2006b) finds that for a reasonable maximum PGV values, no WP-to-WP collision damage is expected to occur.

Therefore, EPRI concludes that DOE has significantly overestimated the PGV and PGA that would occur during reasonable maximum seismic events. This leads to an overestimate of rockfall such that the value of the drip shields in preventing WP damage due to rockfall has been overstated. However, even if significant seismic activity and, hence, rockfall occurs directly onto an unprotected WP, it is EPRI's position that, at most, only a handful of WPs will fail earlier than if drip shields are used.

6.1.5 DOE Overestimated the Likelihood and Rate at which Alloy 22 could Degrade due to Localized Corrosion

It is EPRI's position that DOE has overestimated both the localized corrosion initiation conditions and penetration rate for Alloy 22. Overestimates of these conditions and rates would artificially accentuate the importance of the presence of drip shields.

As described below, DOE conservatively applied a crevice initiation model in two different ways. Crevice initiation was assumed to occur anywhere on the WP surface, even though DOE recognizes crevice initiation will be much more localized:

> Crevices may form on the waste package surface at occluded regions, such as in between the waste package and the emplacement pallet Alloy 22 surfaces and potentially beneath mineral scales, corrosion products, and rocks. It is not expected that the entire waste package surface will be subjected to crevice-like conditions; therefore, application of the crevice repassivation potential model as a criterion for the initiation of localized corrosion to the area subjected to seepage, is conservative. (DOE, 2008b, Section 2.3.6.4.3.1.3)

Furthermore, DOE conservatively assumed there is no critical temperature below which no localized corrosion would occur:

> … The modeling approach did not incorporate a critical temperature below which no localized corrosion would occur, regardless of other conditions in the bulk chemical

> exposure environment. In fact, the empirical rules used to implement the corrosion initiation model (Section 2.3.6.4.4.1) include evaluation of corrosion initiation down to exposure temperatures as low as 20°C. (DOE, 2008b, Section 2.3.6.4.3.1.3)

EPRI (2007b, Section 5.9.5) finds that crevice initiation is highly unlikely – even under aggressive chemical conditions: "…it is unlikely that multiple-salt deliquescent brines could form on WP surfaces in drifts at Yucca Mountain, and, if such brines were to form and be stable for some reason, that they would be incapable of initiating and sustaining localized corrosion of the Alloy 22 outer boundary." Only a small fraction of the possible water chemistries could potentially support localized corrosion. This water accounts for only 1% of all of the possible waters at YM so that, on average, localized corrosion is only possible in 1 out of every 100 realizations in EPRI's WP degradation model (EBSCOM). Initiation in EBSCOM is treated using a threshold temperature for localized corrosion.

Thus, EPRI's opinion is that DOE's assumption that crevice corrosion can occur over the entire WP surface is conservative.

Once crevice corrosion is initiated, DOE then applied a conservative localized corrosion penetration rate. DOE assumes a constant penetration rate with time and also applies a rate for aggressive chemical conditions:

> … a range of potential localized corrosion rates is determined for two highly aggressive environments: (1) 10 wt % $FeCl_3$ test solution (12.7 μm/yr) … and (2) concentrated HCl solutions at elevated temperatures (where passive film is degraded), with corrosion rates between 127 and 1,270 μm/yr. … *The use of an Alloy 22 corrosion rate of 12.7 μm/yr measured in a $FeCl_3$ solution containing about 2.1 m chloride ions at 75°C is a suitable analogue crevice solution for estimating the lower bound for metal dissolution …* because this represents a transpassive corrosion condition. [emphasis added, From DOE, 2008b, Section 2.3.6.4.2.3]

In contrast, EPRI analysis finds that pits will stifle, i.e., crevice corrosion rates will drop to *zero* before the crevice has penetrated the Alloy 22 (EPRI 2004).

Furthermore, DOE implies that crevice corrosion will have only a minor effect on mean dose rates even if the drip shields fail early:

> … although the Alloy 22 localized corrosion abstraction … is part of the TSPA model, there are no modeling cases in which the detailed results of the localized corrosion abstraction result in a dose consequence. … The only modeling case impacted by localized corrosion is the drip shield early failure modeling case, where it is assumed that the waste packages underneath the failed drip shields are failed by localized corrosion. … *Because the occurrence rate is so low for early drip shield failures, this assumption is conservative, but only slightly.* [emphasis added, From DOE, 2008b, Section 2.4.2.3.2.1.2]

Thus, EPRI concludes that DOE has overestimated both the potential for crevice initiation and the localized corrosion rate. Given DOE's overestimations, the longevity of the WPs has been underestimated. This underestimation results in an inappropriately high relative importance of the drip shields to delay onset of localized corrosion.

6.1.6 DOE Neglected Cladding and Inner Stainless Steel Waste Package

Performance

DOE has conservatively assumed that the CSNF cladding will not provide any sort of barrier to the delay or release rate of radionuclides from the UO2 waste form [DOE 2008b, Section 2.4.2.3.2.3.2.3]. Neither has DOE taken credit for the performance of the inner stainless steel canisters within the waste packages or the outer stainless steel shell of the TAD. Without taking any credit for the performance of cladding or the inner stainless steel barriers, the performance of drip shields would seem to be more important than it really is.

EPRI does find that there is sufficient basis for taking credit for the performance of the CSNF cladding in its TSPA model (EPRI, 2000). Available data on the corrosion of zircaloy CSNF cladding were evaluated to derive an estimated cumulative failure curve as a function of time (Figure 6-4). It was assumed that approximately 2% of the cladding was failed prior to emplacement in a repository. After eventual failure of the waste package/EBS, two corrosion modes for the cladding were considered: (1) general corrosion under dry (moist air) conditions, and (2) general corrosion under dripping conditions. At 10,000 years after EBS failure, about 20% of cladding was projected to have failed under dripping conditions and no additional cladding failures were predicted to occur under dry conditions (EPRI, 2000).

Therefore, EPRI concludes that DOE's failure to take credit for the performance of the cladding is overly conservative. Failure to take credit for this additional, available engineered barrier to function as both a barrier and a delay mechanism results in an artificial increase in the relative importance of drip shields.

While it is certain that the stainless steel shells in the waste packages will provide some delay of radionuclide release and reduction of release rates, neither DOE nor EPRI have attempted to quantify this performance. Taking credit for this performance would also diminish the relative importance of the drip shields.

6.1.7 General DOE use of the More Conservative of Available Models

DOE also notes that in general, it uses the more conservative of multiple models that provide a reasonable representation of available data. Several of these conservatisms have caused DOE to underestimate the performance of the EBS components other than the drip shields. These several conservatisms, taken together, represent a significant compounding of each individual conservatism. Because the performance of an entire series of EBS components has been underestimated, together these underestimates have caused DOE to conclude that the addition of drip shields is a necessary component of the EBS.

EPRI disagrees that the drip shields are a necessary component of the EBS. EPRI analyses assuming no drip shields, shown in Figure 6-4b, indicate that the dose rates to the RMEI out to 1,000,000 years after repository closure is still significantly less than the proposed EPA dose limits.

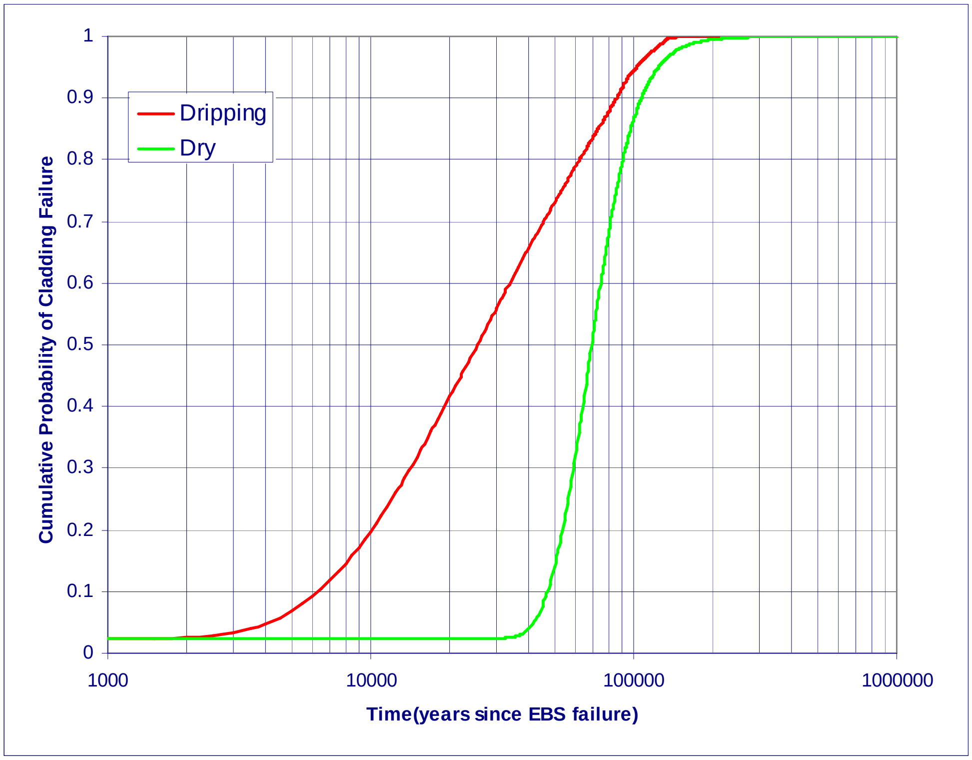

Figure 6 - 4
Derived Cumulative Failure Curves for Zircaloy Cladding (EPRI, 2000)

6.1.7 General DOE use of the More Conservative of Available Models

DOE also notes that in general, it uses the more conservative of multiple models that provide a reasonable representation of available data. Several of these conservatisms have caused DOE to underestimate the performance of the EBS components other than the drip shields. These several conservatisms, taken together, represent a significant compounding of each individual conservatism. Because the performance of an entire series of EBS components has been underestimated, together these underestimates have caused DOE to conclude that the addition of drip shields is a necessary component of the EBS.

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6.1.8 Peak Dose Sensitivity with and without Drip Shields

EPRI performed a TSPA analysis using its IMARC code to compare EPRI's Base Case (drip shields present) and a sensitivity study for which EPRI assumed the drip shields were not present. Figure 6-5 shows the IMARC results for the Base Case (Figure 6-5a) and that for no drip shields (Figure 6-5b). There is a moderate increase in doses at early times associated with the waste package that is assumed to be initially failed owing to manufacturing defects. It is noteworthy that even in the EPRI analyses, the assumption of one initially failed waste package is a conservatism as the expected value of waste package failures from manufacturing defects is significantly below one. The change in peak dose without the presence of the drip shields is negligible, and is still well below the limits established in the proposed EPA/NRC standards.

Based on all the considerations in Section 6.1, EPRI concludes that drip shields are unnecessary.

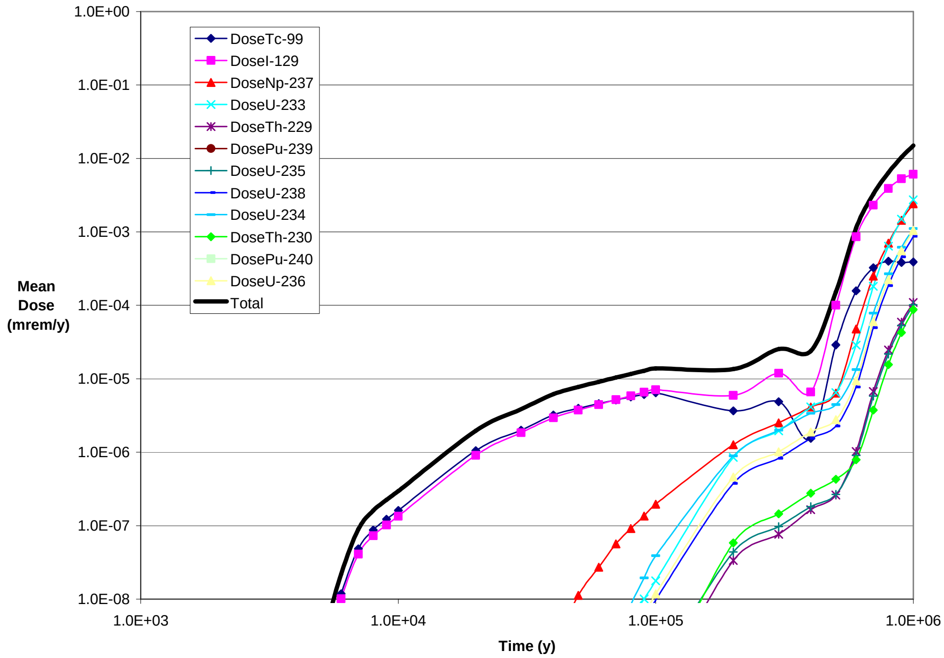
6.2 Impacts of Drip Shield Installation

The YMSEIS (DOE, 2008d) assumes that the annual individual dose associated with installation of the drip shields is 9.75 mrem per year, with a staffing of 10 persons per year, resulting in a total dose of 97.5 person-mrem per year. The repository closure phase is assumed to last for 10 years, although it is not clear from the YMSEIS whether the drip shield installation operations will take place during the entire 10-year operations-closure phase. If drip shield installation takes five years, the total dose would be 487.5 person-mrem. If it takes ten years, the total dose for drip shield installation would be 975 person-mrem. (BSC, 2007)

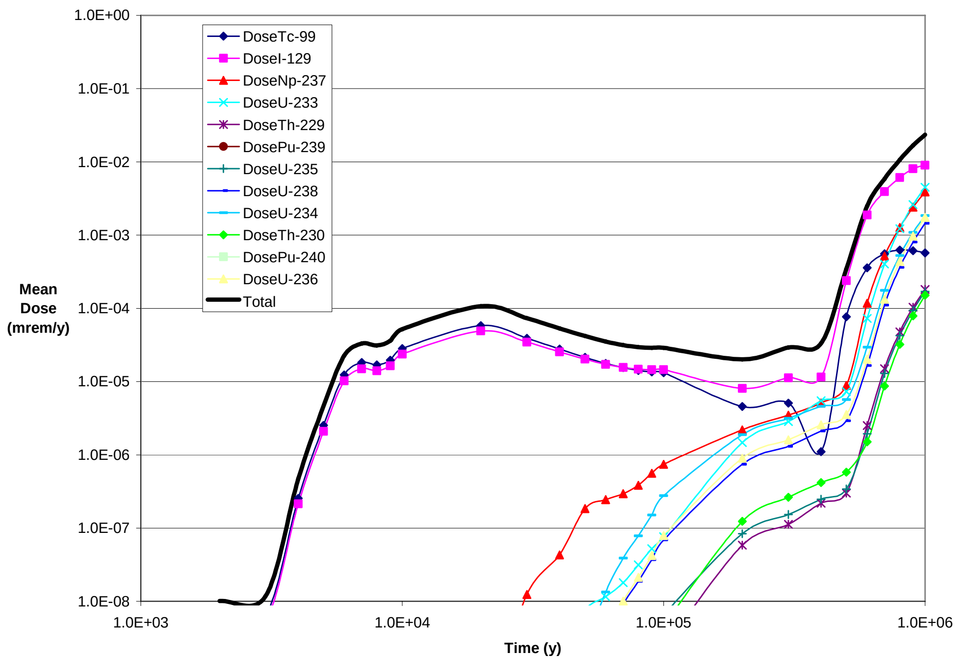
Non-radiological impacts are estimated in a similar fashion. For a five-year period for drip shield installation, the resulting estimates for worker impacts are 4.1 TRC, 2.7 LWC, and 0.009 fatalities. For drip shield installation over the entire ten-year closure period, the estimated non-radiological worker impacts would be 8.2 TRC, 5.4 LWC, and 0.018 fatalities.

Table 6 - 3
Summary of Worker Impacts Associated with Drip Shield Installation

Assumed Duration of Drip Shield Installation (years)	Total Worker Dose (person-mrem)	Non-Radiological Impacts (Cases)
5	487.5	4.1 TRC 2.7 LWC 0.009 fatalities
10	975	8.2 TRC 5.4 LWC 0.018 fatalities



(a)



(b)

Figure 6 - 5
Comparison of EPRI's Base Case (a) and No Drip Shield (b) TSPA Results

By advocating the use of drip shields, DOE is creating substantial resource demands for titanium (Ti), a material of significant strategic importance and of limited domestic availability.[3] DOE estimates that its projected schedule for drip shield manufacture will result in consumption of 22% of present day annual U.S. production of Ti for a limited period of time. Moreover, manufacture of the drip shields incurs occupational risks to involved workers. The YMSEIS estimates that 11,500 drip shields will be used under the Proposed Action. And as a heavy component, the YMSEIS also assumes that 25 drip shields will be shipped per rail car, with a total of 460 shipments. The YMSEIS assumed a shipping distance of 3,464 km, resulting in potential pollution health effect fatalities of 0.028 and vehicle fatalities of 0.036 – or total fatalities of 0.064 associated with the transport of drip shields from manufacturing facilities to the proposed repository (DOE 2008b, Transportation File, Attachment 12, Other materials).

In addition to the fatalities associated with transport of the drip shields, offsite manufacturing of 11,500 drip shields is estimated to require 3.5 million labor hours. The YMSEIS analysis of off-site manufacturing health and safety impacts assumed 9.1 injuries per 100 full-time worker years and 3.29 fatalities per 100,000 worker years. This results in 159 injuries and 0.609 fatalities associated with off-site manufacturing of the drip shields. (DOE 2008b, Offsite Manufacturing File, Attachment A.) These injuries or fatalities could be avoided if there was no need for the manufacture of Drip Shields for placement within the repository.

[3] Although not studied in this report, the resource demand for palladium may also be substantial.

7

DOE HAS OVERDESIGNED PRE-CLOSURE SURFACE FACILITY STRUCTURES FOR SEISMIC RISK AND EFFECT MITIGATION

7.1 Technical Bases

7.1.1 Design of Pre-closure Surface Important to Safety (ITS) Facility Walls is Very Conservative

The facility descriptions in Section 1.2 of the Licensing Application (LA, DOE (2008a)) indicate that the ITS portions of the four main processing structures, the Receipt Facility, Initial Handling Facility, Canister Receipt and Closure Facility, and the Wet Handling Facility (RF, IHF, CRCF and WHF, respectively) are all designed primarily with 4-ft thick external and internal walls. The total length of the walls that will be constructed cannot be currently estimated, as the floor plans for these buildings have been classified "For Official Use Only" (FOUO). However, walls of this thickness require special construction procedures to account for the heat generated during the concrete curing process. In addition, the large volume of the reinforcing bar and concrete required will increase the risk of accidents during construction.

The design basis for the 4-ft thickness appears to be due to seismic loads. Neither radiological safety of protection from aircraft crashes should be the controlling factor for the wall thickness. Per Section 1.6.3.4.1 of DOE (2008b), event sequences due aircraft impact has been screened out, citing "the probability of an aircraft crash is 3×10^{-5} over the preclosure period, which is less than the screening threshold of 10^{-4}. In addition, a procedural safety control on control of aircraft over-flights will be implemented…" It be noted that the fuel will be in shielding casks except during transfer operations. Therefore, the safety benefit of the walls against aircraft crashes is insignificant.

If the design is driven by the need for shielding following accidents, these walls will provide a gamma attenuation in excess of 10^6. This compares to roughly 3-ft concrete thicknesses used in concrete dry storage casks for shielding purposes.

The subsections below examine the design and robustness of the walls against seismic events. The paragraph below summarizes EPRI's opinion based on our review of available documents.

Based on a review of the seismic criteria document in BSC (2007b) and the results documented for the CRCF in BSC (2008), EPRI finds that the HCLPF capacities (High Confidence of Low Probability of Failure) calculated for the ITS structures indicate that these structures are over designed, and wall thicknesses can be reduced while maintaining the required safety levels against seismic failure. The required HCLPF capacity is recommended to be 10% higher than the demand imposed by the 10,000 year return period earthquake or 1.1x0.91g=1.0g (BSC, 2007b, page 48). The HCLPF capacities are to be calculated using the energy dissipation factor of 2.0 corresponding to Limit state A (imminent collapse) given in ASCE/SEI 43-05. BSC (2008), Table 6.2-1, indicates the HCLPF capacity of the CRCF is 1.82g). The 1.82g capacity reported is based on an energy dissipation factor of 1.75, and therefore the capacity corresponding to Limit state A is actually (2.0/1.75)x1.82g=2.1g. This is twice the required capacity of 1.0g, which suggests the thickness of the walls can be reduced while maintaining sufficient seismic margin to easily meet the design requirements. The DOE seismic assessment appears to have recognized this. Recommended refinements in fragility calculations are provided in section B.6 of BSC (2008), but have not been implemented by DOE.

7.1.2 Seismic Design Evaluation

The DOE seismic design basis for surface ITS structures are provided in Table 7-1.

The seismic basis for DBGM-2 (Design Basis Ground Motion -2) are::

- Events with a mean annual probability of exceedance (MAPE) of 5 x 10^{-4} (2,000-year return period), designated as Category 2 events. (0.45g, as shown on Figure 7-1)
- BDBGM (Beyond Design Basis Ground Motion) are events with a MAPE of 10^{-4} (10,000-year return period). (0.91g, as shown on Figure 7-1)

Table 7 - 1
DOE Seismic Bases for Analysis and Design [taken from BSC (2007b), page 11]

Location	SSCs	Seismic Basis[a] for Analysis/Design
Surface	Aging Pads	DBGM-2
	Canister Receipt and Closure Facility	DBGM-2
	Emergency Diesel Generator Facility	DBGM-2[b]
	Initial Handling Facility	DBGM-2
	Receipt Facility	DBGM-2
	Wet Handling Facility	DBGM-2

Figure 7-1 (BSC, 2007a, p. 75) shows the horizontal seismic hazard curve that DOE is using for the YMP. The figure shows the values of horizontal peak ground acceleration (PGA) at 100 Hz applicable to the design of the surface facilities. The critical evaluation for safety purposes is the Beyond Design Basis Ground Motion (BDBGM) event, which is 0.91g for the return period of 10,000 years.

The DOE structure fragilities are provided in Table 7-2, which reproduces Table 6.2.-1 of BSC [2007b] This table shows that the example citing the CRCF used above does not reflect the most robust facility. Both the IHF and the RF can survive a more severe seismic event than the CRCF. It should be noted that the LLW is designed to other criteria, as it will not have contact with CSNF.

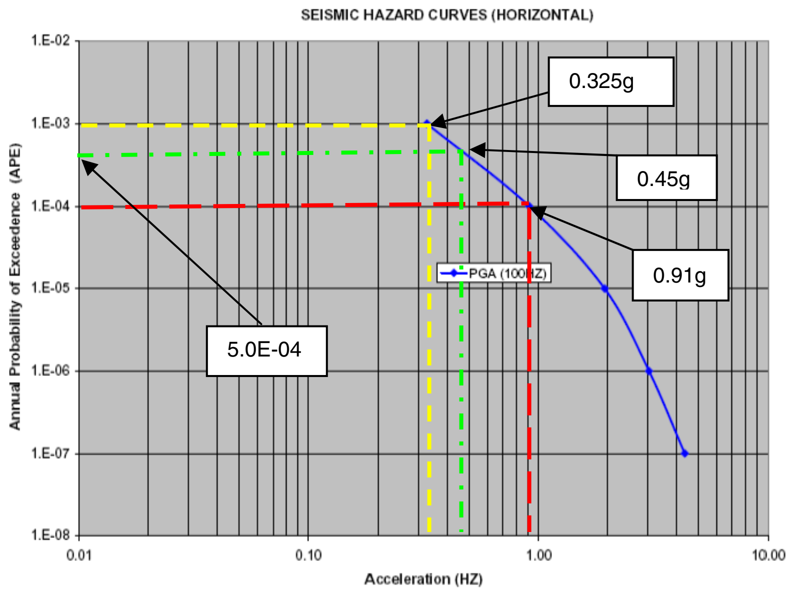


Figure 7 - 1
DOE Horizontal Seismic Hazard Curve [adapted from DOE 2008b, Figure 1.1-89]

Table 7 - 2
DOE Structure Fragilities [BSC, 2007b, Table 6.2-1]

Table 6.2-1. Fragilities for Structures

Structure	Am	βc	HCLPF	Frequency of Failure (/yr)	Basis / Reference
IHF	5.35	0.4	2.11	3.8E-07	Ref. 2.2.33 and Ref. 2.2.32
RF	5.25	0.4	2.07	4.1E-07	Ref. 2.2.72
CRCF	4.61	0.4	1.82	7.8E-07	Ref. 2.2.26
WHF	4.51	0.4	1.78	8.7E-07	Ref. 2.2.54
WHF pool	4.51	0.4	1.78	8.7E-07	Same as WHF building
Aging pad (shear and bending)	2.46	0.4	0.97	9.8E-06	Ref. 2.2.75
Horizontal aging module	4.5	0.4	1.77	8.8E-07	Based on other ITS structures above
LLW Facility	0.89	0.4	0. 35	1.6E-04	IBC, SUG III design (Ref. 2.2.47)

NOTE: CRCF = Canister Receipt and Closure Facility; HCLPF = high confidence of low probability of failure; IHF = Initial Handling Facility; ITS = important to safety; LLW = Low-Level (radioactive) Waste; RF = Receipt Facility; WHF = Wet Handling Facility.

7.2 Impacts of Seismic Over-design

The avoidable occupational health impacts of DOE's ITS surface structure design for seismic risk mitigation are caused by unnecessary construction material production and transportation, and unnecessary on-site construction activities. Appendix C provides a general description of construction-related occupational health impacts per unit time per Full-time Equivalent (FTE) worker. It is difficult to identify what fraction of the total construction health impacts summarized from the YMSEIS in Table C-5 could be avoided if the ITS structures had not been over-designed for seismic risk mitigation. However, since this table indicates that the highest worker risk is during the construction phase (rather than during the operations or closure phases), the avoidable construction risk may be significant.

Specific facility design information is omitted from the publicly available version of the License Application as a result of its designation as "Official Use Only" information. In the absence of this level of detail, EPRI attempted to evaluate the occupational consequences on a more generic, semi-quantitative level using a stylized approach based on the available dimensions for the WHF footprint, typical above-grade height, and wall thickness. EPRI assumed for the purpose of this illustration a WHF facility comprised solely of a rectangular concrete shell (an extremely conservative assumption that neglects interior walls, and contributions from the roof and foundation components). As part of this approach, EPRI ignored the contributions from roof, base mat/pad, and interior walls. The data, assumptions, calculations are described in Section C.3.6.

For a representative structure derived from the description provided in the LA for the WHF, the concrete volume of 438,400 ft^3 was calculated for the 4-foot thick exterior walls occupying an ITS footprint of 385 ft. x 300 ft. For a ready mixed concrete truck capacity of 240 ft^3, this corresponds to a total of 812 truck loads.

Accordingly, for illustration purposes, any unjustified margin resulting from overly conservative treatment of seismic hazards will be reflected in additional use of construction materials and FTEs and the additional burden of occupational risk to workers. For example,

A 10% over-design margin corresponds to 43840 ft^3 or 81 concrete truck loads.

A 25% over-design margin corresponds to 109600 ft^3 or 203 truck loads.

Clearly, any unnecessary and unjustified conservatism in the construction of WHF and other surface pre-closure facilities will result in incremental increases in worker risk due to well-documented occupational hazards. In addition to the often repeated fact that the construction industry is a perennial leader in occupational injury and fatality rates, the Bureau of Labor Statistics has also singled out three specific occupations that exhibited exceptionally high fatality rates in 2005: structural iron and steel workers, truck drivers, and construction laborers.

In lieu of specific occupational risk estimates for the WHF construction, fatality rate data from Table C-9 and injury/fatality rate data from Table C-10 are presented below (Tables 7-3 and 7-4).

The reinforcement of concrete structures to withstand seismic loads directly involves all three of these high-risk occupations for the preparation of appropriate concrete forms, assembly of additional rebar, and pouring of additional concrete. Additional concrete also results in additional truck deliveries that could number in the 100's to 1000's for the case of an over-designed facility. Accordingly, the purposeful over-design (beyond standard engineering margins) for seismic or any other hazard represents unnecessary and unjustified imposition of risk to the involved workers.

Table 7 - 3
Selected occupations with high fatality rates for 2005 (BLS, 2006a)

	Fatalities (per 100,000)
Structural iron and steel workers	55.6
Driver/sales workers and truck drivers	29.1
Construction laborers	22.7

Table 7 - 4
Relevant BLS[a] and DOE[b] Non-radiological Injury and Fatality Rates

Category	**TRC**	**LWC**	**Fatalities**
BLS - construction	6.3	3.4	11.0
BLS - warehousing and storage	8.2	5.4	17.6[c]
BLS - truck transportation	6.1	3.9	17.6[c]
DOE - construction period[b]	2.0	0.86	0.55

[a]BLS, 2006a,b
[b]DOE 2008d, Table 4-16, Section 4.1.7.1
[c]Fatalities for transportation and warehousing category, NAICS code 48-49

It should be noted that DOE's injury and fatality rates are substantially lower that reported by BLS. DOE does not differentiate between specific trades and occupations such as iron workers.

In addition to occupational consequences, the over-design of facilities also consumes significant quantities of materials and resources that would have beneficial uses elsewhere, especially in terms of concrete (cement and aggregate) and rebar (iron/steel).

8 DOE HAS OVERESTIMATED POST-CLOSURE SEISMIC RISK AND EFFECTS

8.1 DOE Overestimated Post-closure Seismic Risk and Effects

As discussed in Section 7.1.4 above, EPRI has determined that DOE has overestimated both the post-closure seismic risk and the effects on repository performance due to seismic activity.

As shown in Figure 8-1, DOE's overestimate of post-closure seismic risk and repository performance effects has caused DOE to find that a major contributor to peak RMEI dose is due to seismic ground motion. While it is not possible to estimate the results DOE would have produced if it had used more reasonable seismic risk and repository performance effects assumptions, it is likely the dose rate estimate for this scenario would be lower, perhaps considerably so. If the igneous intrusion and eruptions scenarios had been eliminated from consideration, as EPRI states is appropriate in Section 5 of this report, and more reasonable estimates of seismic risk and repository performance were used, it is quite possible that the resulting total dose estimates DOE would have derived would be as much as two orders of magnitude lower. If so, this would cause DOE's peak dose results to be fairly similar to the results calculated by EPRI (shown in Figure 8-2).

8.2 Impacts of Post-closure Seismic Risk and Effects

It is unclear what effect DOE's overestimate of post-closure seismic risk and repository performance effects may have on occupational health and safety risk. For example, the DOE overestimates likely have caused DOE to make the TADs more robust than necessary, thereby adding manufacturing complexity. If so, then the additional manufacturing complexity itself may cause an increase in occupational health and safety risk. Furthermore, if the TAD manufacturing process takes longer than a more reasonably designed TAD for more realistic post-closure seismic risk, there could be a delay in moving CSNF from reactor sites to Yucca Mountain.

The occupational health impact due to a one-year delay in the opening of the Yucca Mountain repository has been described in Appendix B and C, and summarized in Table 5-2 of this report.

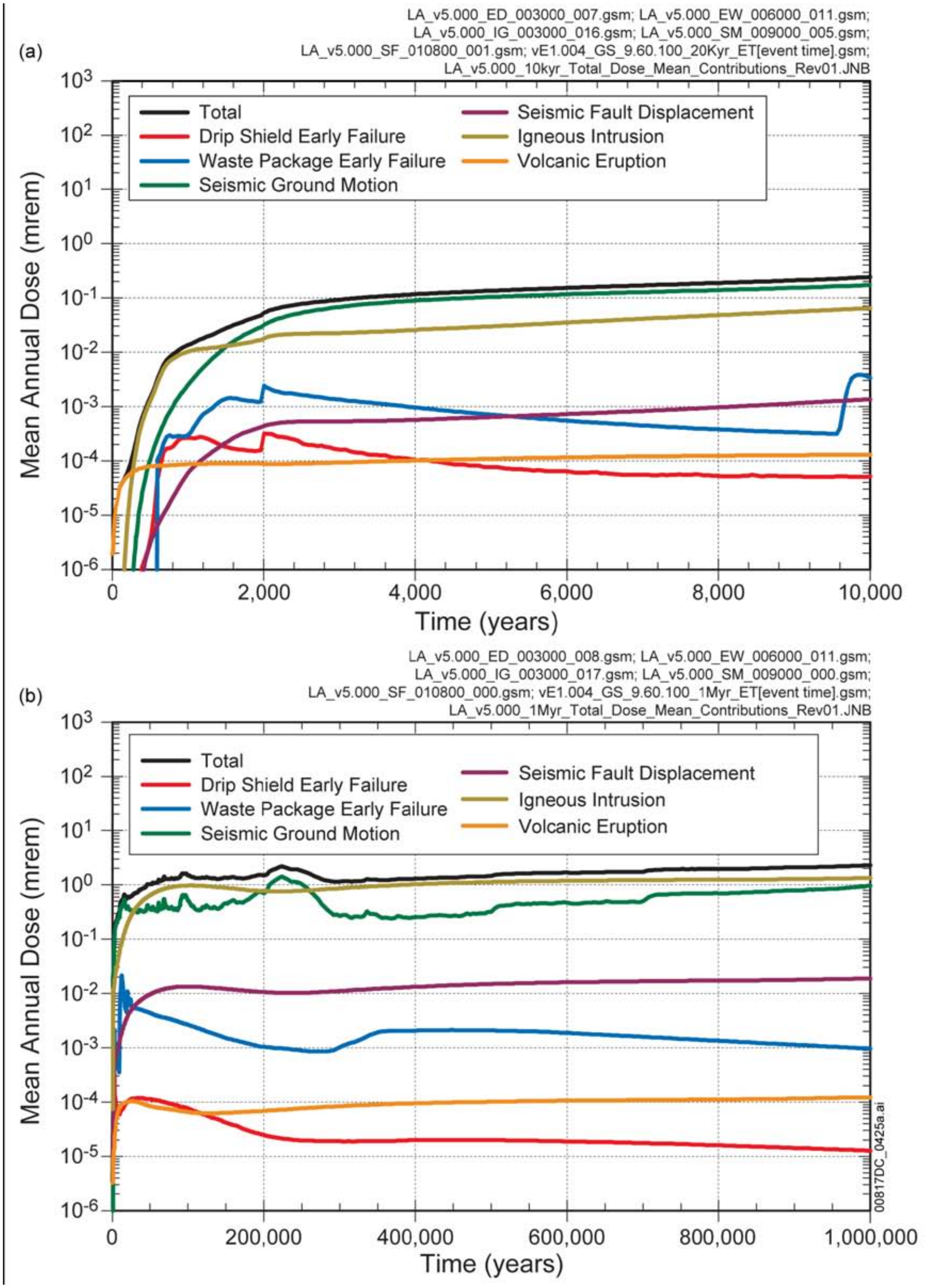


Figure 8 - 1
DOE estimates of RMEI mean annual dose for the 0 to 10,000- (upper figure) and 0 to 1,000,000-year (lower curve) time frames (taken from Figure ES-58 in DOE, 2008b).

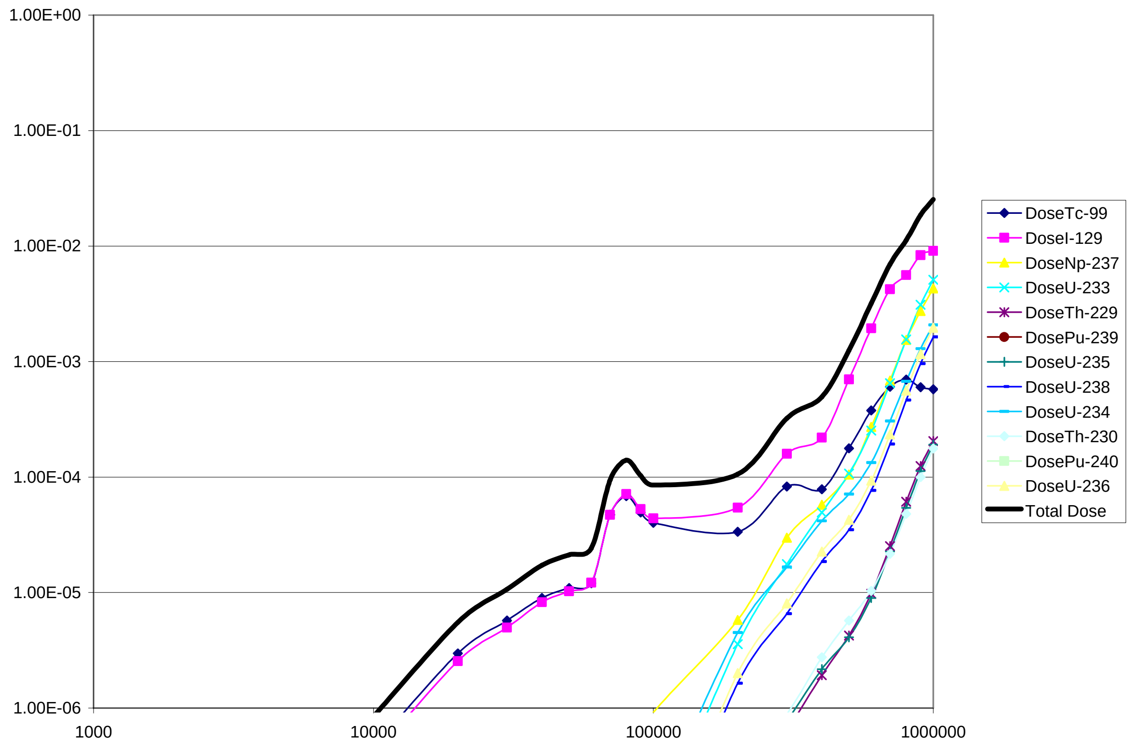


Figure 8 - 2
EPRI CSNF waste package dose results for the Base Case + Seismic Scenarios. Axis units: x: years after repository closure; y: RMEI annual dose rate (mrem).

9

DOE'S APPROACH TO WASTE PACKAGE DESIGN FOR TADS AND CO-DISPOSAL WASTE PACKAGES MAY RESULT IN LICENSING DELAY

With the introduction of the TAD concept for transportation, aging, and disposal of CSNF, DOE has made the TADs more robust than the defense co-disposal waste packages. The Alloy 22 outer shell of the TAD is 25mm while the Alloy 22 shell for the co-disposal waste packages remains at 20mm. Furthermore, the double stainless steel inner canisters in the TADs provides more structural integrity. Thus, DOE concludes: "The CSNF WP is demonstrably more robust [than the co-disposal WP] based on a comparison of the probabilities of damage to WPs with intact internals." (DOE, 2008b, pg. 6.6-7)

DOE notes that the first peak in its RMEI dose estimate, shown in the lower figure in Figure 8-1, is primarily due to failure of the defense co-disposal waste packages. This peak rivals that of the ~1,000,000-year peak caused primarily by the TADs containing CSNF. While EPRI finds significant conservatisms in DOE's TSPA analyses for both the co-disposal and TAD waste packages, DOE's performance assessment results may cause an unnecessary amount of regulatory scrutiny to be placed on the co-disposal waste package behavior. If so, then the repository licensing process may take longer to complete, thereby resulting in a potential delay in the opening of the repository.

The occupational health impact due to a one-year delay in the opening of the Yucca Mountain repository has been described in Appendix B and C, and summarized in Table 5-2 of this report.

10
DOE'S PROPOSED REPOSITORY DESIGN CALLS FOR UNNECESSARILY LARGE SPACING OF DISPOSAL DRIFTS

10.1 Technical Bases

The current repository design uses closely-spaced waste packages (10-cm spacing between each nuclear waste package) in each disposal drift, but with an 81-meter pitch between emplacement drifts. The fundamental rationales of this design include:

- Close spacing within each drift (10 cm) causes each drift to simulate a 'line-load' of radiogenic heating with intense but uniform heating along the entire length of each emplacement drift,
- Radiogenic heating causes localized boiling and removal of water within the emplacement drift and to a limited extent within the surrounding tuff, and
- Extended spacing between drifts (81 meters) and limited extent of boiling around drifts assures that a sub-boiling pillar of tuff rock persists for all time between the neighboring emplacement drifts to allow continuous drainage of any condensate water that may collect above the repository.

This extremely large 81-m pitch is a conservative design, which is relatively space-inefficient compared to other designs that are within rock-mechanical constraints determined by the need for mechanical stability of the drifts (EPRI 2006a, Appendix A). There is no legal or regulatory 'criterion' for such a space-inefficient 81-m pitch between emplacement drifts.

Proposed repository designs as recent as the 2002 Final Environmental Impact Statement or FEIS (DOE, 2002a), however, did not incorporate such a sub-boiling pillar preserved for all time. Instead, it was assumed that the eventual formation of sub-boiling pillars as the rate of radiogenic heating decreased, with lateral water diversion along fractures in tuff, would adequately assure drainage of early-formed condensate water above emplacement drifts that had pitches much smaller than 81-m. For example, a 29-m pitch between emplacement drifts was used in the FEIS design (DOE, 2002a; 2002b) and acceptable repository performance was obtained for such a design.

Furthermore, it should be noted that drift-scale thermal calculations by the USDOE/YMP indicate that the mean average lateral extent of boiling is about 8 m from the drift waters for representative infiltration rates and thermal conductivity (K_{th}) values for tuff (Buscheck et al., 2006). Thus, the 81-m pitch between drifts represents nearly a 4- to 5-fold engineering conservatism compared to the reasonably expected value for lateral extent of boiling, and this 81-m pitch appears considerably larger than is needed to accommodate the expected variability of rock conditions.

10.2 Occupational Health Impacts of an Unnecessarily Wide Disposal Drift Spacing

EPRI agrees that the 81-meter spacing between drifts is unnecessarily wide (EPRI, 2006a; 2007a). Even a drift spacing of half this amount would still provide drainage of groundwater between the drifts. This would approximately halve the required length of the access drifts (as well has halve the repository footprint area, which could impact the probability of igneous activity). *EPRI estimates that reducing the drift spacing to approximately 40 meters would reduce the required length of the access tunnels by approximately four kilometers and the volume of excavated rock by roughly 100,000 cubic meters.*

Appendices B.3.4 and C.3.4 include calculation detail and estimates of the occupational health impacts of unnecessary tunnel excavation. Table 10-1 presents radiological and non-radiological impacts to subsurface workers at Yucca Mountain resulting from the excavation of (potentially unnecessary) four kilometers of access drifts based on estimates for incremental risk per meter of drift excavation.

Table 10 - 1
Additional Worker Dose, Injuries, and Fatalities due to Unnecessary Excavation

Activity	Additional Worker Dose (person-rem)	Additional Worker Injuries and Fatalities
4 km Drift Excavation	48	5.6 TRC 2.4 LWC 0.0015 Fatalities

Although not quantified in this report, there would also be additional, potentially significant occupational risks associated with drift development such as installation of rock support, ventilation equipment, and other subsurface infrastructure.

11
DOE UNDERESTIMATED THE NUMBER OF REQUIRED WET HANDLING FACILITIES

11.1 Potential DOE Wet Handling Facility Throughput Underestimate

DOE's Wet Handling Facility (WHF) is designed to handle all CSNF arriving at the Yucca Mountain site in a container other than TADs. Table 11-1 provides DOE's estimate of the lifetime throughput capacity of the WHF. Based on EPRI's estimates of the number of casks and assemblies that could need to be handled at the Yucca Mountain surface facility, discussed in Section 4.1 of this report, the capacity of the proposed WHF (see Table 11-1) is insufficient to process the anticipated quantity of CSNF that will require processing in that facility.

Table 11 - 1
Wet Handling Facility Design Throughput Capacity over the Pre-closure Period [from DOE (2008b), Table 1.7-5]

Wet Handling Facility	
[Truck] Transportation casks containing uncanistered SNF assemblies (9 BWR or 4 PWR SNF assemblies per cask)	3,775
[Rail] Transportation casks or shielded transfer casks containing a DPC	346
Aging overpacks containing a DPC	346
DPCs (64 BWR or 25 PWR SNF assemblies per canister)	346[c]
SNF assemblies transferred in the pool of the WHF (from an uncanistered-SNF transportation cask or DPC to a staging rack, and from a staging rack to a TAD canister)	66,208[d]
TAD canisters produced at repository (44 BWR or 21 PWR SNF assemblies per canister)	1,165
Aging overpacks or shielded transfer casks containing a TAD canister	1,165

According to the DOE report, "Preliminary Wet Handling Facility Throughput Study", 050-00C-WH00-00200-000-003, ENG.20071102.0019 Informal Study, the WHF is be designed to meet the following throughput criteria:

> The WHF shall be designed to be capable of receiving 230 MTHM per year of bare CSNF from legal weight trucks, over-weight trucks and rail based bare fuel casks, as well as 77 MTHM per year of CSNF in DPCs by rail. In the event that the DOE determines that rail access to the repository site will be unavailable to support system operating conditions and receipt rates, the previous acceptance rates will not apply and will, instead, be based on the availability of truck transportation capability. [050-00C-WH00-00200-000-003, p.13]

The Preliminary Wet Handling Facility Throughput Study estimated the throughput capability of the preliminary WHF design based upon 32 simplifying assumptions, rather than a realistic assessment of anticipated throughput under normal operating conditions. The objective of this throughput estimate was to assist in design development and to provide initial conformance verification that the facility is capable of meeting the assigned processing rates. Results appearing in Table 1 on page 11 of the preliminary throughput assessment are reproduced here as Table 11-2.

Table 11 - 2
Summary of DOE's Proposed CSNF Throughput in the Wet Handling Facility (DOE, 2008b)

Table 1 presents a summary of the throughput model results. For full documentation of throughput model results, refer to Section 5.1.

Table 1. Summary of Throughput Model Results

	Model Results		
Scenario	**TADs Produced[a]**	**Transportation Casks[b]**	**MTHM[c]**
Truck Only	36	191-192	309-315
DPC Only	46-47	44-46	410-418
Mix of Truck and DPC	40-52	61-147	363-464
Small, Med, Large Rail Bare CSNF	54-74	60-138	461-627

Notes: [a] See Table 4 [b] See Table 6 [c] See Table 5

Criterion 3.1.1 requires the WHF to be able to process a combined 307 MTHM per year. The results presented in Table 1 show that the WHF meets the throughput requirement for waste streams containing truck only, mix of truck (bare-fuel) and DPC transportation casks, and DPC only cases.

050-00C-WH00-00200-000-003, p.11

Many of the simplifying assumptions that the preparers of the report acknowledge make their predictions optimistic. Three of the most significant are Assumptions 1, 3 and 32.

Assumption 3.2.1 states, "On demand delivery conditions were used in the throughput model. All inputs, such as loaded transportation casks and new TADs, were available when required. All outputs, such as empty transportation casks, empty DPCs, and loaded TADs, were removed when ready." This assumption requires that all supporting activities external to the WHF be available at all times when the WHF is operational, which is unrealistic. The study made this assumption to limit the scope of the assessment to the WHF, and the authors state that this assumption is suitable for use in only a preliminary engineering study. In making this assumption, they state that it produces an "optimistic", i.e., non-conservative, result.

Assumption 3.2.3 states, "For the purposes of this preliminary throughput study, facility availability was assumed to be 75 percent. The 25 percent non-availability was used to account for routine maintenance and equipment failures, off-normal operations, and recovery time." In essence every potential delay is covered by the 25 percent non-availability.

Assumption 3.2.32 states, "Manpower will be sufficient to support all operational phases based on the WHF operating on the operational work week schedule. This assumption includes sufficient personnel to support activities required to be performed concurrently identified in this throughput." This assumption requires that sufficient personnel be hired, trained, and retained to cover all potential disruptions, such as sickness, vacation, holidays, mandatory training, etc., which is extremely difficult to accomplish.

To account for all potential delays the assessment assumes that the facility will be available 75% of the year.

In presenting the results, the report acknowledges the potential impact of the assumptions on page 10 prior to presenting the results: "The model results are considered optimistic, and per Assumption 3.2.1, outside factors are not represented in the WHF model specifically. While not known in detail, it is anticipated that the outside factors will degrade the performance of the WHF. The primary outside factors include sequencing, the delivery of trucks from truck staging, railcars from rail staging, export of TADs within aging overpacks to either the CRCF or Aging Facilities, delivery of empty TADs, and arrival of site transporter from the Receipt Facility and the Aging Facilities."

Criteria 1.3.1, to which the WHF throughput is being designed, appears to be too low in light of the inventory of dry storage casks currently in dry storage and the number of DPC and dry storage casks that are and continue to be generated by the utilities prior to the availability of the TAD.

Scenarios for Wet Handling Facility CSNF Processing Throughput Needs

EPRI considered three bare fuel, dual-purpose canister (DPC) scenarios that would be shipped to Yucca Mountain by a combination of truck and rail in order to estimate the required number WHFs:

1. DOE's Proposed Action;
2. DOE's Proposed Action except 100% of the DPCs are assumed to be shipped to Yucca Mountain; and
3. EPRI's projected number of casks, canisters and assemblies arriving at Yucca Mountain not in TADs (per EPRI's estimates in Section 4).

EPRI also used these same three scenarios to evaluate the additional processing time required if just one WHF were available.

11.1.1 Number of WHFs Needed to Process the CSNF in 24 Years

Scenario 1: DOE's Proposed Action

According to DOE's Proposed Action, the following numbers of transportation casks and assemblies are anticipated by DOE:

- By rail: 307 DPCs (22,917 assemblies) – 13 DPCs per year
- By truck: 2,650 casks (13,944 assemblies) – 110 casks per year
- 22,428 PWR and 14,433 BWR assemblies for a total of 36,861 assemblies

Comparing to Table 10-1 from Preliminary Wet Handling Facility Throughput Study shown above, one WHF should be sufficient to handle all the casks, DPC canisters, TADs and Aging Overpacks in DOE's Proposed Action.

Scenario 2: 100% of the Projected Number of DPC are Shipped to Yucca Mountain

However, if *all* of the DPCs DOE projects to exist, prior to the widespread use of TADs, are shipped under Proposed Action (during the 24-year pre-closure loading phase DOE proposed), but still assuming the same number of truck casks in DOE's Proposed Action, then the following number of DPCs and truck casks would need to be handled in a WHF:

- By rail: 966 DPCs (37,435 assemblies) – 40 DPCs per year
- By truck: 2650 casks (13,944 assemblies) – 110 casks per year
- 24,940 PWR assemblies and 26,439 BWR assemblies for a total of 51,379 assemblies

This would result in an average of 150 casks being processed through the WHF annually – somewhat more than DOE's estimate of 61 to 147 casks for a mix of truck and DPCs as identified in the Table 1 above, from the Preliminary Wet Handling Facility Throughput Study. Thus, the design capacity of the WHF is not sufficient throughput to handle the total 966 DPCs that DOE has estimated will be loaded for dry storage at reactor sites along with 2,650 truck casks. Depending upon whether one assumes the high or low range of cask throughput (61 to 147 casks) – the facility may need to be expanded by as little as 5% or by more than double the design capacity. Thus, assuming the lower throughput, two wet handling facilities would be needed.

Scenario 3: EPRI's Projected Number of Assemblies not in TADs

The following is a summary of EPRI's projected number of DPCs that will exist at the time TADs enter widespread use in the industry: Using EPRI's 2,375 DPC number

- By rail: 2,375 DPCs. This is based on the following:
- 28,820 assemblies in the 845 canisters already loaded
- 67,550 assemblies in the 1,530 canisters projected to be loaded.
- Total assemblies: 96,370 composed of 44,525 PWR and 51,845 BWR assemblies

In EPRI's estimate of the total number of DPCs that may be loaded at reactor sites by 2020, EPRI assumed that the sites that DOE identified as shipping by truck would actually ship CSNF via large capacity rail casks. Some of these sites would load DPCs for at-reactor storage and are included in EPRI's estimate that as many as 2,155 DPCs and an additional 220 storage-only canisters may be loaded at reactor sites through 2020 and shipped to the repository. If these packages must be unloaded in the WHF, this would result in a total of 2,375 canister systems being unloaded during the 24-years of the Proposed Action, or an average of 99 DPCs or canisters per year. The total CSNF assembly throughput for the WHF would be 96,370 assemblies, or an average of 4015 assemblies per year.

As noted in the Preliminary Wet Handling Facility Throughput Study, in a DPC-only scenario, a total of 44-46 DPC transportation casks could be unloaded annually at the WHF. This is less than half of the average of 99 DPCs that would have to be handled using EPRI's estimate of 2,155 DPCs or canister systems. Thus, if utilities load as many as 2,375 DPCs and the DPCs are transported to the repository during the 24-year Proposed Acton, it appears that the WHF throughput would not be adequate to handle these additional packages and that the WHF capacity would have to be doubled. DOE has not assessed the worker impacts associated with construction and operation of an additional WHF.

11.1.2 Additional Processing Time if Just One WHF were Available

Alternatively, for Scenarios 2 and 3 one WHF could be adequate, but it would take a longer period of time to process all the casks, canisters, and assemblies arriving in non-TADs.

If just one WHF were required to handle the amount of CSNF not in TADs described in Scenarios 2 and 3 above, a rough estimate of the additional amount of time is as follows. It is assumed that the maximum number of bare fuel, DPCs, and assemblies DOE can handle in a single WHF is based on Table 11-1. Furthermore, it is assumed that it takes 24 years for DOE to handle this amount of CSNF.

Scenario 2

Cask-limited:

For this estimate, it is assumed that DOE can process the same number of casks whether by truck or rail. However, it is likely that it will take a longer period of time to process a DPC arriving by rail that bare fuel arriving by truck. This is because there are extra steps involved in processing a DPC compared to processing bare fuel.

- Number of casks requiring processing: 966 (DPCs) + 2650 (truck) = 3616
- Number of casks DOE can process in one WHF in 24 years: 346 (DPCs) + 3775 (truck) = 4121

Therefore, based on the conservative assumption that it takes the same amount of time to process 4 PWR or 9 BWR assemblies arriving as bare fuel in a truck cask, and >24 PWR or >40 BWR assemblies arriving in a DPC, one WHF could process all the assemblies in Scenario 2 in 24 years. Since it is more likely that it will take longer to process a DPC than a truck cask, it is likely that it will take somewhat more than 24 years to process the CSNF arriving as described in this scenario.

Assembly-limited:

- Number of assemblies requiring processing: 61,669
- Number of assemblies that one WHF can process in 24 years: 36,861
- Number of years to process 61,669 assemblies: 24 X (61,669/36,861) = 40 years

Therefore, it would require an additional 16 years to process the additional amount of CSNF in this scenario, assuming the WHF processing time is somewhat insensitive to whether the assembly being processed is from a PWR or a BWR.

Scenario 3

Cask-limited:

- Number of casks requiring processing: 2375 (DPCs) + 2650 (truck) = 5025
- Number of casks DOE can process in one WHF in 24 years: 346 (DPCs) + 3775 (truck) = 4121

- Amount of time to process: 24 years X (5025/4121) = 29 years

Therefore, if processing time in the WHF is limited by the number of casks that can be handled, then it would take a minimum of five additional years to process the required number of casks. In reality, it is likely to take considerable more than five additional years as this estimate assumes the same amount of processing time for a DPC arriving in a rail cask and 4 to 9 assemblies arriving bare in a truck cask.

Assembly-limited:

- Number of assemblies requiring processing: 96,370
- Number of assemblies that one WHF can process in 24 years: 36,861
- Amount of time to process: 24 years X (96,370/36,861) = 63 years

Therefore, if processing time in the WHF is limited by the number of assemblies that can be handled, then it would take a on the order of 39 additional years to process the required number of assemblies.

The incremental occupational health risk due to a one-year delay in the availability of Yucca Mountain is described in Section 5-2. Estimates of the potential delay if one WHF is available is between 0 and 39 years, although the delay may be longer than 39 years it if takes longer to process one, large DPC compared to one small truck cask, which is likely.

11.2 Impacts of an Insufficient Number of Wet Handling Facilities

For Scenarios 2 and 3, it would be necessary to institute some combination of increasing the number of WHFs and decreasing the amount of time required for processing the necessary quantity of casks, canisters, and assemblies in a single WHF. Any solution would delay the ability of Yucca Mountain to receive CSNF in any container other than a TAD. While not discussed in any detail in this section, either alternative would also incur additional cost.

If additional WHF were required, it is possible that DOE would need to build them over several years as DOE may be funding-limited. Given the considerable cost of constructing such a facility and the need to obtain requisite funding, the additional time required to complete construction of additional WHFs could be significant. Construction of additional WHF(s) will also cause an increase in occupational health risk due to the necessary construction and material requirements.

If it takes longer to process an additional amount of CSNF in a single WHF, then the utilities would incur both additional costs and occupational health risk as it would become necessary for the utilities to keep non-TAD containerized CSNF in storage at their sites for a longer period of time. The occupational health impact due to a one-year delay in the opening of the Yucca Mountain repository has been described in Appendix B and C, and summarized in Table 5-2 of this report.

11.2.1 Occupational Health Impacts Associated with the Construction of Additional Waste Handling Facilities

As described in Section 7.2, EPRI chose to evaluate the occupational health impacts on more generic, semi-quantitative level by calculating concrete volumes required for construction of a stylized Yucca Mountain surface facility based on the available dimensions for the Waste Handling Facility footprint, the typical above-grade height, and wall thickness. Accordingly, the example is applicable to this discussion as well. This illustration assumes a WHF facility comprised solely of a rectangular concrete shell -- an extremely conservative assumption that neglects interior walls, and contributions from the roof and foundation components. The data, assumptions, calculations are described in Section C.3.6.

As described earlier, this simplistic approach was necessitated by the lack of data provided in the License Application due to the designation of design information as "Official Use Only" in the public document.

For a representative structure derived from the description provided in the LA for the WHF, the concrete volume of 438,400 ft^3 was calculated for the 4-foot thick exterior walls occupying an ITS footprint of 385 ft. x 300 ft. For a ready mixed concrete truck capacity of 240 ft^3, this corresponds to a total of 812 truck loads.

The construction of one or more additional WHFs represents a major undertaking in terms of costs, materials, and workforce. Along with the significant requirement for construction related workers come some of the highest occupational risks of any industry. As highlighted in Section 7.2 and Table C. 9, the Bureau of Labor Statistics singled out three specific occupational subcategories, structural iron and steel workers, truck drivers, and construction laborers, associated with exceptionally high fatality rates and would be comprise the majority of the workforce for WHF construction.

In lieu of specific occupational risk estimates for the WHF construction, fatality rate data from Table C-9 and injury/fatality rate data from Table C-10 are presented below (Tables 11-3 and 11-4).

Table 11 - 3
Selected occupations with high fatality rates for 2005 (BLS, 2006a)

Occupation	Fatalities (per 100,000)
Structural iron and steel workers	55.6
Driver/sales workers and truck drivers	29.1
Construction laborers	22.7

The reinforcement of concrete structures to withstand seismic loads directly involves all three of these high-risk occupations for the preparation of appropriate concrete forms, assembly of additional rebar, and pouring of additional concrete. Additional concrete also results in additional truck deliveries that could number in the 100's to 1000's for the case of an over-designed facility. Accordingly, the purposeful over-design (beyond standard engineering margins) for seismic or any other hazard represents unnecessary and unjustified imposition of risk to the involved workers.

Table 11 - 4
Relevant BLS[a] and DOE[b] Non-radiological Injury and Fatality Rates

Category	TRC	LWC	Fatalities

BLS - construction	6.3	3.4	11.0
BLS - warehousing and storage	8.2	5.4	17.6[c]
BLS - truck transportation	6.1	3.9	17.6[c]
DOE - construction period[b]	2.0	0.86	0.55

[a]BLS, 2006a,b
[b]DOE 2008, YMSEIS, Table 4-16), Section 4.1.7.1
[c]Fatalities for transportation and warehousing category, NAICS code 48-49

11.2.2 Occupational Health Risk Increase Caused by Additional Time to Process CSNF in One WHF

As described in Section 10.1.2, the additional processing time if just one WHF were available would range between zero and perhaps over 39 years. Appendices B and C include estimates of the occupational health risk associated with a one-year delay in the initiation of CSNF shipments from the reactor sites to Yucca Mountain. These numbers would need to be multiplied by a range of 0 to •39 to provide a rough estimate of the additional occupational health risk due to this delay.

11.2.3 Economic Impacts of Additional WHF Construction

Based on a DOE cost estimate contained in a 2007 DOE budget projection for expenditures from FY2009-FY2023 (DOE 2007), the costs for construction of the Initial Handling Facility ("IHF" which would handle canistered naval reactor SNF and DHLW) and the WHF were estimated to be $615 million. EPRI conservatively assumed that both of these facilities would have equal cost, although it should be noted that the WHF has more complex handling operations and would be expected to have a higher cost than the IHF. This results in an estimated cost to construct additional WHF of $307.5 million.

12

DOE HAS OVERESTIMATED POST-CLOSURE DOSE TO THE PUBLIC DUE TO CONSERVATISM IN REPOSITORY PERFORMANCE ASSESSMENTS

12.1 Technical Bases

There are multiple conservatisms in DOE's repository performance assessment that result in an overestimate of post-closure dose to the pubic. DOE (2008b), Section 1,8 describes these conservatisms. DOE notes the following about the models incorporated into its TSPA-LA:

> The submodels incorporated into the TSPA-LA Model are representations of the repository system. The guiding principles during the development of these submodels were to: (1) ensure that representations were not optimistic (i.e., leading to an underestimation of the dose results), and (2) incorporate all included FEPs. Although these representations were developed to be as realistic as possible, some conservative (reasonable and technically defensible based on supporting analyses) representations were required for complete development of the TSPA-LA. Model. [DOE (2008b), Section 1.8]

These conservatisms include, for example:

1. Overestimate of the importance of colloid-aided radionuclide transport to the biosphere;
2. No credit has been taken by DOE during the post-closure period for the integrity of the rock support system. Given the robust design of the rock support systems, it is likely that this system will continue to perform for potentially a significant amount of time after the repository is closed. This could provide additional protection from rockfall to the underlying engineered barrier system (EBS) components during the early period of highest rock stresses and highest radionuclide activity.
3. No credit is taken for the degradation rate of the stainless steel TAD canister or the inner stainless steel layer of the disposal overpack. Again, if credit were taken for these stainless steel layers, release of radionuclides from the repository would be further delayed;
4. Overestimate of the amount of carbon-14 that would be transported downstream;
5. Overestimate of neptunium solubility, an key actinide for the long-term repository performance.

12.1.1 Colloids

Radionuclides that are retarded in natural systems due to sorption to soil/rock surfaces and/or low aqueous solubilities are potentially subject to rapid transport in the subsurface due to mobile colloid phases. Such facilitated transport processes are especially important for strongly sorbing and low solubility actinides such as Pu.

EPRI has conducted a thorough review of the properties of relevant colloids and the mechanisms by which the different classes of colloids could conceivable operate to enhance the mobility of otherwise immobile radionuclides to the RMEI at the compliance location [EPRI, 2006, # 1013440]. In order for colloid-facilitated transport to play a significant role in the dose to the RMEI, several conditions must exist simultaneously, including the following major ones:

- Colloids must form in sufficiently large numbers to provide sufficient surface area for transport of the inventory of radionuclides;
- Colloids must remain stable for the relevant distance to the RMEI (kilometers) and timeframe (10^4 years);
- Colloids must not be subject to significant reversible or irreversible filtration by the geologic media.

EPRI has determined that none of these conditions will be met for the relevant timeframes and physical scales. Accordingly, it is appropriate to screen colloidal transport out of performance assessment modeling as a dose-significant process.

Moreover, DOE recently replaced mild steel with stainless steel inserts in the proposed standardized TAD canister; in doing so, DOE has also eliminated the potential for formation of iron-oxide/ hydroxide based colloids.

By choosing not to screen facilitated colloidal transport out as relevant process, DOE adds unnecessary complexity into an already complex modeling environment and introduces another layer of conservatism.

12.1.2 Rock Support System Integrity

The rock support system DOE proposes to use is likely to last longer than the time of repository closure. EPRI has not yet studied the issue of long-term rock support integrity, but assuming it does last even a few additional decades, this would be well into the period of maximum EBS temperatures. A generally sound rock support system during this period could prevent any significant amount of rockfall to occur. Given the relatively low relative humidity during the period of the highest temperatures after repository closure degradation of the rock support system via corrosion would likely remain low.

Preventing significant rockfall through the peak temperature period after repository closure could help to reduce damage to the underlying drip shields and waste packages, and could reduce the amount of groundwater seepage into the drifts.

12.1.3. Degradation of Stainless Steel Components of Waste Package

As discussed in Section 6.1.6 above, neglecting the potential structural and radionuclide migration mitigation performance of the outer stainless steel shell of the TAD and the inner stainless steel shell of the waste package is conservative. If DOE had considered these two potential EBS barriers, then DOE would have found improved structural resistance to seismic activity and rockfall, and reduced radionuclide migration rates out of the waste package after waste package failure.

12.1.4. Carbon-14

The DOE approach to C-14 in the waste form, near-field, and far-field is overly conservative, resulting in C-14 representing the second highest dose contributor for the early period of performance -- on the order of 0.04 mrem per year at 10,000 years (DOE, 2008a). Only Tc-99 yields a higher dose, 0.1 mrem/yr, at 10,000 years. Similarly, C-14 also ranks no. 2 in dose contribution for the first 10,000 years for the nominal and seismic ground motion scenarios as a result of DOE's overly conservative approach. In TSPA performance margin analyses, DOE reports C-14 to be the third highest dose contributor for the first 10,000-year period (DOE, 2008a, Vol. III, App. C, p. C-96, 2008).

The factors contributing to C-14's prominent role in early dose include: high solubility in a carbonate/bicarbonate form (CO_3^{2-}/ HCO_3^-), non-sorption, and relatively long half-life with respect to the 10,000-year timeframe. C-14 is treated similarly to Tc-99 and I-129 as a high-solubility, non-sorbing radioisotope (DOE, 2008a, Vol III., p. 8.1-8). No mention appears to be made of C-14 exchange with naturally occurring carbon in groundwater and air in unsaturated tuffs that would lead to substantial evolution of C-14 as a gas prior to transport to RMEI location.

DOE reports that release rates of C-14 from the waste package/waste form, along with Tc-99 and I-129, are limited only by waste form degradation rate, rate and extent of water ingress into WP, and mass transport out of WP (DOE 2008a, Vol III., p. 8.1-8). DOE further claims that C-14 will be transported to the RMEI at the same rate as groundwater (i.e., as a conservative tracer), and will not be subject to retardation or losses other than radioactive decay. Again, this indicates that DOE dose not consider any well-established gas-exchange reactions (occurring over relatively short time frames, of days to weeks, with respect to transport through the unsaturated zone), evaporation, weathering, isotopic fractionation, or precipitation reactions, which would serve to deplete C-14 concentrations in water exiting the engineered barrier system or result in C-14 incorporation into existing and prevalent carbonate minerals within the unsaturated and saturated zones (Langmuir, 1997; Stumm and Morgan, 1981).

Moreover, carbon (as C-14) generated in spent fuel via neutron activation of nitrogen impurities in fuel and hardware components is expected to be in a reduced chemical form (graphite) because of the reducing conditions prevailing during reactor operations. Graphite, as a common material used in such consumer items as pencils and lubricants, does not readily oxidize into carbonate at atmospheric pressure and expected repository temperatures. DOE, however, conservatively assumes that all of the initial C-14 embedded in the fuel matrix and hardware immediately oxidizes to form a highly soluble carbonate or bicarbonate species when contacted by groundwater. By foregoing known and well-understood geochemical reactions and kinetic constraints, DOE (2007) conservatively considers C-14 to be instantaneously released as a 'highly soluble' radioelement.

In the unlikely event that C-14 is released as a soluble carbonate/bi carbonate species, it is important to note that the typical groundwaters in Yucca Mountain tuffs are close to saturation with respect to calcite, a condition confirmed by the prevalence of calcite in fractured tuffs (Paces et al., 2001). For the five cited reference groundwaters (Table 12-1), the calculated mean solubility concentration, with 2-sigma standard deviation, for calcite is $10^{-3.08\pm0.78}$ moles/L. The actual concentration of C-14 in equilibrium with calcite would be further lowered by consideration of normalizing the calcite solubility value by the relative ratio of trace C-14 to the total mass of all naturally occurring carbon isotopes (C-12 and C-13), the so-called isotopic mass fraction.

Accordingly, , if it is conservatively assumed that all of the C-14 can be oxidized and mobilized from spent fuel as carbonate/ bi-carbonate species, the prevailing geochemical conditions at Yucca Mountain would impose rather low values to the range in possible C-14 concentration. For calcite, a 'high' solubility value = 5.0 E-3 moles/L, the 'mid' solubility value = 8.3 E–4 moles/L, and the 'low' solubility value = 1.4 E–4 moles/L are adopted as the reasonably expected solubility values from Table 12-1 compositions. These values would have to be, in turn, reduced by the extremely small mass-fraction of C-14 compared to all carbon isotopes.

Table 12 - 1
Compositions of Representative Yucca Mountain Waters (from Table 6.2-1 of BSC 2003)

Porewater ID	**W0**	**W5**	**W4**	**W6**	**W7**
Lithostratigraphic Unit	Tptpmn	Tptpul (base)	Tptpll	Tptpll	Tptpul
Temperature (°C)	25	25	25	25	25
pH	8.3	7.6	7.4	7.9	8.0
Na^+ (mg/L)	61.5	39.0	130.0	84.0	57.0
K^+ (mg/L)	8.0	7.6	10.6	7.9	10.3
Ca^{2+} (mg/L)	101.0	94.0	82.0	56.0	120.0
Mg^{2+} (mg/L)	17.0	18.1	5.3	0.9	19.3
SiO_2(*aq*) (mg/L)	70.5	42.0	48.0	50.0	49.0
Cl^- (mg/L)	117.0	21.0	26.0	23.0	54.0
SO_4^{2-} (mg/L)	116.0	36.0	39.0	10.0	78.0
HCO_3^- (mg/L; calc)[1]	200.0	395.0	515.0	335.0	412.0
NO_3^- (mg/L)	6.5	2.6	4.2	17.0	6.1
F^- (mg/L)	0.9	3.4	6.0	2.5	4.8

1- Total aqueous carbonate as HCO_3^- (mg/L), calculated from charge balance.

The remaining C-14 in groundwater would be further attenuated by previously mentioned processes such as gas-water exchange, weathering reactions with alumino-silicate minerals, and evaporation.

From a performance margin viewpoint that for the ambient environmental conditions prevailing at the time of initial container failures (1 atmosphere pressure, temperature below 96˚C), if carbon-14 is present as graphite, this form of carbon can remain chemically inert for geological time scales. Furthermore, even if the reduced C-14 becomes oxidized to and is transported as soluble carbonate/bicarbonate species from the near-field of a repository, numerous well-established and naturally evident processes would act to attenuate a significant fraction of C-14 dissolved in groundwater during transit in the unsaturated and saturated zones. All of these factors provide additional performance margins to attenuate or retard the release of C-14 and are not accounted for in DOE's evaluation of Yucca Mountain performance. The net result of these physical-chemical partitioning processes would be a substantial reduction or retardation of the C-14 inventory that would be transported to the RMEI location.

12.1.5 Neptunium Solubility (EPRI, 2005c)

Performance assessment modeling indicates that, after 10,000 years, neptunium-237 (Np-237) and its decay products are dominant contributors to RMEI dose. Because of its long half-life (2.14×10^6 years), the peak dose from Np-237 at the compliance point scales proportionally with the solubility limit for Np. Therefore, a realistic determination of Np solubility behavior in the proposed repository is important for reasonable performance assessments and determination of regulatory compliance for Yucca Mountain.

Previously, the U.S. Department of Energy (DOE) has identified and evaluated three conceptual models to define the maximum concentration of Np at the surface of dissolving spent nuclear fuel (Chen et al., 2002; DOE, 2003):

A *base-case conceptual model*, in which it is conservatively assumption that maximum Np concentrations are limited by the solubility of crystalline Np_2O_5(cr). This Np(V) phase has a solubility of about 10^{-5} M (2.4 mg/L) Np in repository groundwaters (cf. Friese et al., 2004).

A *first alternative conceptual model* that assumes that maximum Np concentrations are determined by the solubility of the Np(IV) solid phase NpO_2 (cr) in the same oxidized groundwaters that were assumed for the base-case model (DOE, 2003). There is evidence that NpO_2(cr) is thermodynamically more stable than Np_2O_5(cr) in the repository (Roberts et al., 2003). The DOE's modeled solubility of NpO_2(cr) is about 1.2 log units (a factor of 17) lower than that of Np_2O_5(cr) (DOE, 2003).

The License Application apparently assumes a combination of the base-case and first alternative conceptual models for Np solubility control. The "Dissolved Concentration Limits of Elements with Radioactive Isotopes" report (Sandia, September 2007, ANL-WIS-MD-000010 Rev 06, DOC 20070918.0010), which provides the data used in the License Application, notes (Sandia, 2007, pages 6-66 to 6-67) that both NpO_2 (a Np (IV) phase) and $NaNpO_2CO_3$ (a Np (V) phase) are considered as solubility-controlling phases inside failed waste packages in which reducing materials (e.g., fuel or steel) are still present, whereas Np_2O_5 and $NaNpO_2CO_3$ (both Np (V) phases as in the *base-case conceptual model*) are assumed if all reducing material is corroded within a failed waste package.

A *second alternative conceptual model* previously identified by the DOE, also described as the *secondary phase neptunium solubility model* (DOE, 2003), assumes that maximum Np concentrations are determined by precipitation of the Np from spent fuel dissolution in solid solution with major secondary uranium minerals. The DOE did not adopt this model in the LA because it was not considered sufficiently supported by experimental evidence (Sandia, 2007, page 6-67). The DOE has previously recognized (DOE, 2003), however, that Np concentrations predicted with this secondary phase neptunium solubility model are in excellent agreement with the concentration of Np released by dissolution of spent fuel, a value which is typically in the range of 10^{-8} to 10^{-10} M, whereas Np concentrations predicted using the base case model or first alternative conceptual model are 3 or more orders of magnitude higher (i.e., more conservative) than experimental evidence.

Based on a review of available published studies presented in this report, EPRI believes that DOE's base case assumption that Np_2O_5(cr) solubility defines maximum possible Np concentrations at Yucca Mountain is unrealistically conservative for the following reasons:

- Pure Np phases have never been observed to precipitate in spent fuel leaching experiments (DOE, 2003). There is no evidence that Np concentrations from the leaching of spent fuel will ever be high enough to result in the precipitation of pure Np(V) phases such as Np_2O_5(cr).
- Thermodynamic databases developed by the DOE (Kaszuba and Runde, 1999; DOE, 2000a), and independently by international groups (Lemire et al., 2001; Guillaumont et al., 2003), indicate that NpO_2(cr) is probably more stable than Np_2O_5(cr) under all repository conditions.
- Laboratory experiments at 90°C and above in oxidized waters have precipitated NpO_2(cr) (Finch, 2001; DOE, 2001; Roberts et al., 2003), suggesting that Np(V) phases such as Np_2O_5(cr) are metastable and, with time, will convert to more thermodynamically stable and less soluble NpO_2(cr) in the repository.
- In experiments most closely simulating the heterogeneous conditions expected during the dissolution of spent fuel in the repository, the Np/U ratio of the leachates is the same as the Np/U ratio of the fuel, and Np concentrations do not increase with time relative to uranium concentrations as secondary uranyl minerals are formed (DOE, 2003). This confirms active uptake and incorporation (co-precipitation) of trace Np into secondary uranyl minerals at approximately the same Np/U ratio as was present in the spent fuel. Resultant Np(V) concentrations can be expected to be extremely low ($<10^{-7}$ to 10^{-9} M) and controlled by the solubility of secondary uranyl minerals and the mass fraction of Np incorporated in those minerals.

Based on these results, it is EPRI's position that, of the three models considered by the DOE, the second alternative conceptual model, the secondary phase neptunium solubility model, is the most realistic and technically defensible to evaluate the long-term release behavior of Np from a repository at Yucca Mountain. The other conceptual models based on the formation and solubility of pure Np-solids are considered to be unrealistic and conservatively bounding.

There is another factor why Np releases from a repository at Yucca Mountain can be expected to be low, providing even more evidence of the conservatism of the base case model. Combined sorption and reduction of Np (V) to Np (IV) can also be expected in groundwater migrating beneath the repository via matrix flow through vitric layers in the tuffs of the Calico Hills Formation[4]. A number of researchers have shown the tendency for Np(V) to be adsorbed by tuff minerals such as magnetite (and probably also ilmenite) that contain Fe(II), with reduction of Np(V) and its retention as less soluble Np(IV) species (Nakata et al., 2002; 2003).

Based on these multiple lines of evidence and reasoning, EPRI concludes that Np concentrations released from a repository at Yucca Mountain will be controlled at values below 10^{-7} M by co-precipitation in secondary uranyl minerals in the near field, and by reduction and sorption as Np(IV) in underlying tuff formations. To purposefully adopt an excessively conservative alternative conceptual model for Np solubility imposes an unwarranted perception of potentially higher doses resulting from a repository at Yucca Mountain than is reasonably supported by data from the DOE and independent international scientific peer groups.

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[4] Assuming its composition is similar to that of the overlying Topopah Spring Tuff as described by Peterman and Cloke (2002), the Calico Hills contains the Fe(II)- bearing minerals magnetite, ilmenite and pyrite at 0.19, 0.18 and 0.09 average weight percent, respectively.

12.2 Potential Impacts

Because DOE's multiple conservatisms lead DOE to overestimate dose rates to the RMEI, the repository system design may be more robust than a repository design based on a different design based on more reasonable assumptions and data inputs to DOE's dose assessment calculations. This could lead to increased time requirements for the design and/or construction of the associated facilities.

A secondary issue is that DOE's conservatisms make the dose estimates appear as if there is only a limited amount of margin below the proposed EPA and NRC dose limits. The more limited the margin between the calculated performance and the established regulatory limits, the greater the potential for increased regulatory scrutiny. Such scrutiny might result in extension of the regulatory process and/or increased litigation regarding any conclusions reached by the regulator.

As discussed in Section 5.2 and Appendices B and C., any delays in the licensing, construction, and operation of the repository places additional radiological and non-radiological risk burdens on workers at the utility sites due to the need to construct additional ISFSI capacity; to extend and/or expand inspection and maintenance programs for existing ISFSI facilities at operating plants

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A

EXISTING AND PROJECTED QUANTITIES OF COMMERCIAL SPENT NUCLEAR FUEL CANISTERS

A.1 Evaluation of Commercial Spent Nuclear Fuel Packaging Scenarios

The Transportation, Aging, and Disposal canister is the only recognized disposable canister for commercial spent nuclear fuel in DOE's license application. The TAD reflects an evolution of an earlier standardized disposal package. The TAD capacity is relatively small with respect to many commercially available canister designs, accommodating only 21 PWR or 44 BWR assemblies. DOE's proposed action for the design and operation of Yucca Mountain accommodates a limited amount of CSNF arriving at the repository in DPCs and other non-TAD-packaging. DOE proposes a baseline of up to 10% of non-TAD CSNF and also evaluates an alternative scenario for up to 25% of inbound non-TAD packaged CSNF [DOE SEIS, 2008]. The consequences of DOE's approach to disposal canister design, repository design, and operations cascade throughout the repository system and extend out to the nuclear utilities, workers, and the general public. Accordingly, understanding the various quantities of DPCs, TADs, and other containerized forms of CSNF is central to evaluating the impacts of DOE decisions relative to the storage, transport, and disposal of CSNF.

EPRI analyses suggest that many of the existing dual-purpose canisters (DPCs) used by the nuclear industry could be safely transported, aged, and disposed of at Yucca Mountain (EPRI, 2008a). Currently licensed DPCs hold approximately 1.14 to 1.55 times as much spent nuclear fuel as do the proposed TADs. Thus, using the proposed TAD size instead of DPCs or larger capacity TADs will result in a larger number of canisters being loaded at nuclear utility sites, transported to Yucca Mountain, potentially aged, and then disposed.

DOE also assumes that SNF from seven commercial nuclear power plants as well as two national laboratories would be transported to Yucca Mountain utilizing truck casks with capacities of 4 PWR assemblies or 9 BWR assemblies. All of the commercial nuclear power plant sites that DOE identifies as using truck casks have plans to or are expected to load large rail-capable DPCs for on-site storage of CSNF. Therefore, in addition to evaluating the impacts associated with DOE's assumed TAD capacity, EPRI also evaluated the impacts to workers associated with the transport of CSNF in truck casks rather than in DPCs or large capacity TADs.

Table A-1, below, provides a summary of the types of packages that DOE assumes will be used to transport CSNF to the Yucca Mountain repository under the 70,000 MTU base case compared to transportation cases identified by EPRI. DOE's Final Supplemental Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada (YMSEIS) assumed that a total of 6,499 TADs, 307 DPCs, and 2,650 truck casks would be loaded with CSNF and transported to Yucca Mountain under the Proposed Action (70,000 MTU repository capacity). In addition, EPRI considers two alternative scenarios, EPRI Case 1 and Case 2, which are described as followes:

- EPRI Case 1 assumes that the sites that DOE identified as loading and transporting 21P/44B TADs instead load larger capacity TADs; a small number of previously loaded DPCs are transported; and that the truck sites identified in the YMSEIS (DOE, 2008a) ship CSNF by truck. This results in the shipment of 4,591 larger TAD packages, 307 DPCs and 2,650 truck casks. EPRI Case 1 is conservative since all of the sites identified by DOE as truck sites have plans or are expected to load large-capacity DPCs for on-site storage.
- EPRI Case 2 assumes that sites identified as loading and transporting 21P/44B TADs instead load larger capacity TADs and that commercial reactor sites designated as truck sites also load larger capacity TADs instead of truck casks. This results in the shipment of 4,928 larger TAD packages, 307 DPCs, and 2 truck casks.

Table A - 1
Estimated Reduction in Number of CSNF Packages Loaded and Transported Associated with Use of Larger TAD Designs

Package Type	DOE YMSEIS	EPRI – Case 1 With Truck Casks	EPRI – Case 2 Minimal Truck Casks
TAD 21P/44B	6,499		
Large Capacity TAD 24P/32P,61B,68B	0	4,591	4,928
DPC	307	307	307
Truck	2,650	2,650	0
Total Casks Shipped	9,456	7,548	5,239

A.2 Projections for Quantities of Dual-Purpose Canisters Loaded at Reactor Sites

The YMSEIS assumes that a total of 307 DPCs and storage-only canister-based systems would be shipped to the repository and unloaded at the repository under the 70,000 MTU repository case and that a total of 966 DPCs would be shipped to the repository and unloaded at the repository if the full MTU of CSF is assumed. (DOE 2008c, Transportation File, Trans data_Summary.xls)

As of May 2008, approximately 625 DPCs had been loaded into ISFSIs for on-site storage at commercial nuclear power plant sites. EPRI has projected that an additional 1,530 DPCs could be loaded at reactor sites between 2008 and 2020. Thus, a total of 2,155 DPCs could be loaded at reactor sites through 2020. EPRI's projection of DPCs loaded through 2020 assumes that nuclear operating companies continue to load DPCs rather than TAD canisters for on-site storage through that date, although it is possible that companies would begin loading TAD canisters at an earlier date if they are available.

In order to estimate the number of additional DPCs loaded through 2020, EPRI projected CSNF discharges for all currently operating nuclear power plants. Average annual spent nuclear fuel discharges are expected to be in the range of 2,100 to 2,300 MTU per year through 2020. Using current and planned CSNF storage pool capacities and projected CSNF discharges, EPRI estimated that approximately 1,700 MTU of dry storage capacity would be needed at nuclear power plant sites annually through 2020. This projection of additional on-site storage assumes that all U.S. licensed nuclear power plants continue to operate through the end of their 60-year extended licenses; that lifetime capacity factors average approximately 90%; and that average discharge fuel burnups gradually increase to 58,000 MWD/MTU for PWRs and 46,400 MWD/MTU for BWRs.

In estimating the number of DPCs loaded through 2020, EPRI assumed:

- Plants with existing ISFSIs that are loading CSNF into metal dual-purpose casks would continue to do so through 2020.
- Plants with existing ISFSIs would continue to load CSNF into packages with similar capacities through approximately 2013.
    - Plants that are now loading 24-PWR DPCs with approximately 10 MTU per DPC, would continue to do so through 2013.
    - Plants that are currently loading 32-PWR or 61/68-BWR DPCs, with approximately 13 MTU per DPC, would continue to do so through 2013.
- Plants with new ISFSIs would load high capacity DPCs (32-PWR or 61/68 BWR.
- From approximately 2014 forward, EPRI assumed that all CSNF would be loaded into higher capacity DPCs at existing ISFSIs and new ISFSIs (except at those sites currently loading CSNF into metal dual-purpose casks as noted in the first bullet, above).

As shown in Table A-2, EPRI estimates that utilities could load as many as 2,155 DPCs at reactor sites through 2020. Utilities have also loaded 220 canister-based storage-only dry storage systems – the YMSEIS assumes that some of these canisters would be transported to the repository for repackaging at the repository. Thus, EPRI estimates that as many as 2,375 DPCs and canister based systems could in use for storage of CSNF by 2020.

EPRI also projects that as many as 135 dual-purpose metal casks could be in storage at reactor sites by 2020. In addition, approximately 101 metal dry storage casks or other storage-only systems have been loaded for dry storage at reactor sites.

Table A - 2
Estimated Dry Storage Systems Loaded at Nuclear Power Plant Sites Through 2020

Package Type	Number of Packages Loaded
Storage-Only Canister Systems	220
Dual-Purpose Canister Systems	2,155
Dual-Purpose Metal Casks	135
Storage Only Metal Casks	101

Table A - 3
Estimated Number of DPCs (and Other Non-TAD Canisters) for Receipt at Yucca Mountain

Estimate	Number of DPCs for Receipt at Yucca Mountain
YM SEIS baseline	307
YM SEIS high DPC	966
EPRI	2375

A.3 Projections for Quantities of TAD Canisters Loaded at Reactor Sites and Yucca Mountain

The YMSEIS assumed that a total of 7,400 TADs would be used for CSNF disposal under the proposed action (DOE 2008a, Table 4-32). As noted in Appendix A.1, the YMSEIS assumes that a total of 6,499 TADs are loaded with CSNF at reactor sites, leaving a total of 901 TADs to be loaded with commercial SNF that is shipped in the 307 DPCs and 2,650 truck casks. Under EPRI Case 1, a total of 4,591 higher capacity TADs are assumed to be loaded at nuclear power plant sites. If the CSNF shipped to the repository in DPCs and truck casks are repackaged at the repository into higher capacity TAD packages (32P, 68B), EPRI estimates that 489 packages would need to be loaded at the repository. Under EPRI Case 2, a total of 4,928 higher capacity TADs are assumed to be loaded at reactor sites. Under this scenario, there were two truck casks containing CSNF, which is assumed to be transferred to a single TAD canister at the repository.

Table A - 4
Estimated Number of TADs Loaded at Reactors Versus Repository for Different Scenarios

Scenario	Number of TADs		
	Loaded at Reactor Sites	Loaded at Yucca Mountain	Total
DOE YMSEIS (70,000 MTU)	6,499	901	7400
EPRI Case 1	4,591	489	5080
EPRI Case 2	4,928	1	4929

A.4 References

DOE 2008a. Yucca Mountain Repository License Application. U.S. Department of Energy, Office of Civilian Radioactive Waste Management, DOE/RW-0573, Rev. 0, June 2008

DOE 2008b. Final Supplemental Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada, DOE/EIS-0250F-S1, LSN #: DEN001593669.

DOE 2008c. Yucca Mountain Supplemental Environmental Impact Statement: Calculation Packages in Support of Final Supplemental Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada, DOE/EIS-0250F-S1.

EPRI 2008. Feasibility of Direct Disposal of Dual-Purpose Canisters in a High-Level Waste Repository. EPRI, Palo Alto, CA: 2008. 1010851.

B
RADIOLOGICAL IMPACTS

B.1 Radiological Impacts at Reactor Sites

Radiological impacts at reactor sites include worker doses associated with canister/cask loading, unloading, and handling activities as well as doses associated with ISFSI operations and maintenance, surveillance activities and additional ISFSI construction as discussed in more detail below.

B.1.1 Radiological Impacts Associated with Cask Loading and Handling

The YMSEIS assumed that workers at commercial nuclear power plant sites would incur radiological risk associated with the loading and handling of packages for transport of SNF as summarized in Table B-1. DOE's estimated worker doses at nuclear power plants included the following:

- 400 person-mrem per large capacity rail cask loaded and transferred to dry storage (this applies to TADs, DPCs, or bare-fuel rail casks) (DOE 2008a, Table G-2)
- 432 person-mrem per truck cask loaded (DOE 2008a, Table G-2)
- 663 person-mrem per package transferred from dry storage to rail cask. The YMSEIS assumed that all TADs loaded would be transferred to dry storage at reactor sites prior to transport by rail to Yucca Mountain. (DOE 2008a Table G-2; DOE 2008b,Transportation File, Attachment_02_Loading, loading_impacts.xls, CI_summary_rad worksheet).

Table B - 1
Doses to Workers At Reactor Sites Associated with Cask Loading and Handling Operations

Activity	Worker Dose (person-mrem/cask)
Canister/Cask Loading Operations ▪ TADs or large rail casks ▪ Truck Casks	 ▪ 400 ▪ 432
Cask Transfer from ISFSI to Rail Cask ▪ DPC ▪ TAD	 ▪ 663 ▪ 663
Cask Unloading Operations ▪ Storage Only Systems ▪ DPC and Dual Purpose Casks	 ▪ 260 ▪ 260

In the Proposed Action in the YMSEIS, DOE did not calculate the impacts associated with unloading storage-only dry storage systems or DPCs used for on-site for repackaging into TAD canisters, rail casks, or truck casks. The YMSEIS assumed that no DPCs or storage-only systems would be unloaded during the Proposed Action. If DPCs or storage-only systems needed to be unloaded, one could estimate the dose by using the same worker dose estimates that the YMSEIS used for unloading DPCs at the repository surface facilities. The YMSEIS estimated that the radiological dose associated with unloading DPCs at the Yucca Mountain repository's Wet Handling Facility would be nominally 260 person-mrem per cask (assuming a collective dose of 13 person-rem/year and 50 casks per year at the Wet Handling Facility). (DOE 2008b, Radiological Health and Safety File, Attachment 1, Worker Tables_D9_D10).

As shown in Table B-2, utilizing the worker dose assumptions for cask loading and handling operations identified in Table B-1, EPRI calculated the impact associated with DOE's decision to utilize the 21P/44B TAD canisters for transport of CSNF to Yucca Mountain rather than utilizing large capacity TADs or DPCs. Table A-1 describes the number of packages assumed for the doses calculated in Table B-2. Compared to ERPI Case 1 assumptions in which larger capacity TADs are loaded at reactor sites for transport to Yucca Mountain, DOE's decision to utilize the 21P/44B TAD design rather than a large capacity TAD would increase worker doses associated with cask loading operations by 2,028 person-rem over the 24 years associated with transport of CSNF to the repository. Compared to EPRI Case 2 assumptions in which larger capacity TADs are loaded and a minimal number of truck casks are assumed for transport of CSNF currently stored at national laboratories, DOE's decision to utilize the 21P/44B TAD design and to ship CSNF from reactor sites using truck casks rather than large capacity rail casks would increase worker doses associated with cask loading operations by 2,813 person-rem over the 24 years associated with transport of CSNF to the repository.

Table B - 2
Estimated Worker Dose Associated with Cask Loading and Handling

	Total Worker Dose Associated with Cask Loading and Handling (person-rem)		
Package Type	**DOE YMSEIS**	**EPRI Case 1**	**EPRI Case 2**
TAD 21P/44B	6,908	0	0
Large Capacity TAD 24P/32P,61B,68B	0	4,880	5,238
DPC	203	203	203
Truck	1,145	1145	2
Total Worker Dose	8,256	6,228	5,443
% Dose Reduction		25%	34%

B.1.2 Radiological Impacts Associated with ISFSI Operation and Maintenance

As shown in Table B-3, below, in addition to the worker dose associated with loading and handling of packages for transport, the YMSEIS assumed that workers would incur the following doses associated with the dry storage of CSNF at 75 reactor sites for 20 years:

- 120 person-mrem per site per year for annual inspection/security surveillance (DOE 2008b, Transportation File, Attachment_02_Loading, loading_impacts.xls, CI_summary_rad worksheet)

- 1,500 person-mrem per site per year for annual maintenance (DOE 2008b, Transportation File, Attachment_02_Loading, loading_impacts.xls, CI_summary_rad worksheet)

B.1.3 Radiological Impacts Associated with ISFSI Expansion and Construction

While the YMSEIS and its associated calculational package did not calculate the additional radiological risk associated with additional construction at reactor site Independent Spent Fuel Storage Facilities (ISFSI), documentation associated with DOE's No Action Alternative did evaluate these impacts. If additional ISFSI construction is required while there is already CSNF in dry storage, DOE's No Action Alternative assumed that there would be an additional 170 person-mrem per additional cask loaded as shown in Table B-3. (Jason 1999, Rollins 1998).

Table B - 3
Doses to Workers at Reactor Sites Associated with ISFSI Operations, Maintenance and Construction

Activity	Unit Impact
ISFSI Operation and Maintenance	
(person-mrem per year per site)	
Inspection and security surveillance	120
Annual maintenance	1,500
Additional ISFSI Construction	
(person-mrem per additional canister stored)	170

B.1.4 Radiological Impacts Associated with Cask Unloading at Reactor Sites

As noted above, the Proposed Action in the YMSEIS does not calculate any impacts associated with unloading storage-only dry storage systems or DPCs used for on-site for repackaging into TAD canisters, rail casks, or truck casks. The YMSIES assumed that no DPCs or storage-only systems would be unloaded at reactor sites during the Proposed Action.

Accordingly, the YMSEIS does not calculate the worker dose associated with unloading CSNF in dry storage at reactor sites for repackaging prior to shipment to Yucca Mountain. However, it is conceivable that at point in the future, some of these packages would need to be unloaded at reactor sites for transfer into TADs. Should this activity become necessary EPRI calculates that industry workers would incur a dose of 260 person-mrem per package unloaded.

Table B-4 summarizes the potential worker dose associated with unloading, at reactor sites, the dry storage packages identified in Table A-2. The YMSEIS does not address CSNF stored in dual-purpose metal casks or storage-only metal casks in terms of transport or repackaging. For this inventory, EPRI estimated a cumulative worker dose of 35 person-rem associated with unloading dual-purpose metal casks and 26 person-rem associated with unloading storage-only metal casks at reactor sites for repackaging prior to transport to the repository. As noted earlier, the YMSEIS assumes that 307 to 966 DPCs and/or storage-only canister systems will be transported to the repository for repackaging under the Proposed Action 70,000 MTU repository scenario and the Model 1 repository scenario, respectively. EPRI estimates that as many as 2,375 DPCs and storage-only canisters could be in use at reactor sites by 2020.

Table B - 4
Estimated Worker Dose Associated with Unloading Dry Storage Systems at Nuclear Power Plant Sites

Package Type	Number of Packages Unloaded	Estimated Worker Dose Unloading Operations (person-rem)
Storage-Only Canister Systems	220	57
Dual-Purpose Canister (DPC) Systems	2,155	560
Dual-Purpose Metal Casks	135	35
Storage Only Metal Casks	101	26

Although not considered by DOE as part of the LA, if these systems had to be unloaded at reactor sites for repackaging prior to transport, EPRI estimates a worker dose of 57 person-rem and 560 person-rem for unloading storage-only canister systems and DPCs, respectively.

B.2 Radiological Impacts to Workers During Incident-Free Transport

The YMSEIS estimates the impacts for maximally exposed workers associated for incident free transport of SNF and HLW to the repository. (DOE 2008a, Table 6-5). Shipment escorts and inspectors were assumed to receive the highest radiation doses due to their proximity to the casks. The YMSEIS made the following assumptions regarding incident free worker dose:

- Escorts, rail inspectors 0.5 rem per year
- Rail yard crew member 0.1 rem per year
- Truck driver 0.5 rem per year
- Truck inspector 0.2 rem per year

In order to calculate the collective incident free transportation impacts to workers, the YMSEIS utilized unit risk factors to provide an estimate of the radiation doses from transport of one shipment or container of radioactive material over a unit distance of travel in a given population density zone. Unit risk factors can provide an estimate of the radiation dose from one container or shipment being stopped at a location such as a rail yard or the radiation dose from one container or shipment passing a train stopped at a siding. The unit risk factors were combined with the cask, shipment, population density, and distance data to calculate collective dose. Unit Risk Factors for workers, used to calculate worker collective dose, are identified below.

Worker Unit Risk Factors – Incident Free Transportation, CSNF (DOE 2008b, Transportation File)

- Workers at stops
  - Enroute 6.03 x 10^{-6} person-rem/km
  - Near generator sites: 1.87 x 10^{-2} person-rem
- Workers during train assembly: 2.74 x 10^{-2} person-rem
- Security escorts
  - Rural: 2.02 x 10^{-5} person-rem/km
  - Suburban 3.23 x 10^{-5} person-rem/km
  - Urban 5.39 x 10^{-5} person-rem/km
- Security escorts:
  - At stops enroute: 9.36 x 10^{-6} person-rem/km
  - At stops near generator sites: 2.60 x 10^{-3} person-rem

As shown in Table B-5, the YMSEIS calculated the collective radiological impacts to transportation workers associated with incident free transportation of CSNF to the repository. (DOE 2008a, Table 6-4). The YMSEIS calculated a 2,833 total rail shipments (assuming three casks per train) for CSNF, DOE and Navy SNF, and HLW; and 2,650 truck shipments. As summarized in Table B-5, under EPRI Case 1 there would be an estimated 2,074 rail shipments, assuming three casks per train, and 2,650 truck shipments. Under EPRI Case 2 there would be an estimated 2,186 rail shipments (assuming 3 casks per train) and 2 truck shipments. The estimated rail shipments in the DOE YMSEIS, EPRI Case 1 and EPRI Case 2 include shipments of CSNF, DOE and Navy SNF, and DOE HLW.

The YMSEIS estimated the collective incident free radiation dose to workers associated with the transport of SNF and HLW by rail was 4,700 person-rem and by truck was 880 person-rem. For 2,833 rail shipments the average is 1.7 person-rem per rail shipment. For 2,650 truck shipments, the average dose per shipment is 0.3 person-rem.

As shown in Table B-5, if rail shipments of CSNF utilized higher capacity casks than the 21P/44B TAD design as assumed in EPRI Case 1, the estimated worker dose would be 4,326 person-rem – *a reduction of 1,174 person-rem compared to the worker dose calculated in the YMSEIS*. If rail shipments of CSNF utilized higher capacity TADs and the truck sites identified in the YMSEIS instead shipped by higher capacity TADs, the estimated worker dose would be 3,717 person-rem – *a reduction of 1,783 person-rem compared to the worker dose calculated in the YMSEIS*.

Table B - 5
Estimated Worker Dose Associated with Transport of SNF to Yucca Mountain

Scenario	Number of Shipments	Estimated Worker Dose (person-rem)
DOE YMSEIS (70,000 MTU)		
▪ **Rail Shipments (3 casks per train)**	2,833	4,700
▪ **Truck Shipments**	2,650	800
EPRI Case 1		
▪ **Rail Shipments (3 casks per train)**	2,074	3,526
▪ **Truck Shipments**	2,650	800
EPRI Case 2		
▪ **Rail Shipments (3 casks per train)**	2186	3,716
▪ **Truck Shipments**	4	1

B.3 Radiological Impacts to Workers at Yucca Mountain

B.3.1 Radiological Impacts Associated with Receipt, Handling, and Aging of CSNF

The YMSEIS assumed that workers at the Yucca Mountain repository would incur radiological risk associated with the receipt, handling, aging and permanent disposal operations. As shown in Table B-6, the YMSEIS included surface facility dose rates for each of the surface facility operations.

Table B - 6
Estimated Worker Dose Associated with Unloading Dry Storage Systems at Yucca Mountain Surface Facilities

Facility	Waste Type	Nominal Number Casks per Year	Annual Dose (Person-Rem)	Dose per Cask (Person-mrem/Cask)
Cask Receipt Security Station	***All Packages***	365	10	27
Receipt Facility	***All Packages***	210	36	172
Canister Receipt and Closure Facilities	***TAD, DOE SNF, DOE HLW***	216	27	125
Wet Handling Facility	***Bare CSNF and DPC***	50	13	260
Aging Facility	***DPC and TAD***	135	6	44

As shown in Table B-7, utilizing the worker dose assumptions for the receipt facilities identified in Table B-6 and the number of packages handled summarized in Table A-1, EPRI calculated the worker dose impacts associated with handling CSNF at the Yucca Mountain suface facilities.. Compared to ERPI Case 1 assumptions in which larger capacity TADs are loaded at reactor sites for transport to Yucca Mountain, DOE's decision to utilize the 21P/44B TAD design rather than a large capacity TAD would increase worker doses associated with cask handling operations at the Yucca Mountain surface facilities by 701 person-rem over the 24-years of the Proposed Action. Compared to EPRI Case 2 assumptions in which larger capacity TADs are loaded and a minimal number of truck casks are assumed for transport of CSNF currently stored at national laboratories, DOE's decision to utilize the 21P/44B TAD design and to ship CSNF from reactor sites using truck casks rather than large capacity rail casks would increase worker doses associated with cask handling operations at the Yucca Mountain surface facilities by 1,792 person-rem over the 24 years associated with the Proposed Action.

Table B - 7
Estimated Worker Dose Associated with Handling CSNF at Yucca Mountain

Facility	DOE YMSEIS		EPRI – Case 1		EPRI – Case 2	
	Number Packages	Worker Dose (Person-Rem)	Number Packages	Worker Dose (Person-Rem)	Number Packages	Worker Dose (Person-Rem)
Cask Receipt Security Station	9456	255	7548	204	5239	141
Receipt Facility	9456	1,626	7548	1,298	5239	901
Canister Receipt & Closure Facilities TAD	6499	812	4591	574	4928	616
Wet Handling Facility						
DPC	307	80	307	80	307	80
Truck	2,650	689	2,650	689	4	1
Aging Facility						
TAD	6499	286	4591	202	4928	217
DPC	307	14	307	14	307	14
TOTAL DOSE		3,762		3,061		1,970

B.3.2 Radiological Impacts to Workers at Yucca Mountain Associated with Unloading Additional Dual-Purpose Canisters

As discussed in Section A.2, the YMSEIS assumes that 307 to 966 DPCs and/or storage-only canister systems will be transported to the repository for repackaging under the 70,000 MTU repository scenario and the full MTU scenario, respectively. EPRI estimates that as many as

2,375 DPCs and storage-only canisters could be in use at reactor sites by 2020. EPRI contends that it is possible that some of these DPCs and storage-only canisters may be able to be placed in a waste package for direct disposal, without repackaging.

If these systems were transported to Yucca Mountain and unloaded, rather than being placed in waste packages for direct disposal, *a net additional worker dose of 135 person-mrem per package (260 person-rem – 125 person rem from Table B-6) would be incurred to unload the additional DPCs or disposal canisters* (Table B-8). Accordingly, this same dose also represents the potential dose avoided per canister if direct disposal of DPCs or other canisters were incorporated into DOE operations.

Table B - 8
Estimated Net Worker Dose Associated with Unloading DPCs (and Storage Only Canisters) at Yucca Mountain

Scenario	Number of DPCs for Receipt at Yucca Mountain	Worker Dose for DPC Unloading (person-rem)
YM SEIS	307	41
EPRI Case 1	966	130
EPRI Case 2	2375	320

B.3.3 Radiological Impacts Associated with Management of Empty DPCs as Low-Level Radioactive Waste

Every DPC canister unloaded (for transfer of CSNF inventory to TAD) will generate a significant quantity of low-level radioactive waste (LLW) that needs to managed and disposed of safely, incurring additional doses and non-radiological risks to workers at the point of fuel transfer (utility or Yucca Mountain receipt facility). DOE assumes each DPC represents 10.6 m^3 in LLW volume (DOE, 2008b, waste file, June 2008, filename: CalcPkg_Waste1_Attach2_MG_9.19.07.xls.) For the proposed action in the YMSEIS, DOE explicitly considers 307 DPCs for receipt at Yucca Mountain, which would need disposal as LLW once the CSNF inventories were transferred to TADs. The corresponding volume of LLW from DPC disposal in this case would be 3,254 m^3 or 4% of the total projected LLRW volume of 74,000 m^3 (Table 4-31, DOE, 2008a) to be processed for the project. However, in the YMSEIS, DOE also provides an upper bound estimate on DPC derived LLRW based on the ultimate disposal of 920 DPCs, which correspond to a total volume of 9,800 m^3 (DOE, 2008a) or 13% of total LLW volume for the pre-closure period of the project.

DOE estimates that LLW facility operations will result in worker doses of 9 person-rem per year (DOE, 2008b waste file, June 2008, Rad H&S File, Attachment 1, Worker Tables). In the absence of more specific information such as activity of individual waste streams, it is difficult to attribute dose to DPCs. For simplicity, EPRI assumes that doses from the management of DPC wastes are proportional to waste volumes. Accordingly, a 4 – 13% DPC waste volume range would yield 0.36 – 1.2 person-rem/year dose to workers from DPC waste management activities. Collective worker dose associated with the Low-Level Waste Facility during the operations period are estimated by DOE to be 310 person-rem (Table D-9; DOE 2008a, Vol II., Appendix D, p. D-22). For a total LLW volume of 74,000 m3, this corresponds to 4.2 x 10^{-3} person-rem/m^3.

EPRI has independently calculated that there could be as many as 2,375 DPCs and other non-TAD canister-based systems loaded for storage of CSNF at reactor sites. If these DPCs are unloaded at Yucca Mountain for transfer of CSNF to TAD canisters, there will be corresponding increases in the volume of LLW requiring disposal and worker dose. Assuming a LLW volume of 10.6 m^3 for each DPC discarded, the corresponding volume of LLW would be 25,175 m^3, a 15,375 m^3 increase over the DPC volume assumed in the YMSEIS (DOE, 2008a). Using the 4.2 x 10^{-3} person-rem/m^3 unit dose calculated above, disposal of one DPC yields a unit dose of 0.045 person-rem. Likewise, disposal of a total of 2,375 DPCs would result in additional collective worker doses of 65 person-rem over the operations phase of the project. Disposal of the empty DPCs offsite along with other LLW would impose radiological risks to workers at commercial facilities. Relevant occupational doses associated with commercial low-level radioactive waste management are reported by NRC up through the year 1998 in NUREG-0713.[5] Worker doses of 0.1 rem/year appear representative for the most recent NRC data.

Table B - 9
Estimated Worker Dose Associated with Management of Empty DPCs (and Storage Only Canisters) as Low-Level Radioactive Waste

Scenario	Number of DPCs for Receipt at Yucca Mountain	Worker Dose for DPC Management as LLRW (person-rem)
YM SEIS (baseline)	307	14
YM SEIS High DPC Estimate	966	43
EPRI DPC Estimate	2375	110

The current DOE proposed approach does not call for unloading of DPCs at generator sites; however, it in conceivable that such a burden could be shifted to utilities and other ISFSI operators. Any LLW management activities resulting from the unloading of DPCs at the plant site would result in comparable (non-trivial) doses to workers at the generator end of the supply chain.

B.3.4 Radiological Impacts Associated with Additional Subsurface Construction Resulting from the Exclusive Use of Low Capacity TAD Canisters for CSNF

DOE's decision to use relatively low capacity 21P/44B TAD canisters will require the excavation of more emplacement drifts and associated access drifts than if higher capacity TADs and/or DPCs were accommodated in the proposed action. Accordingly, each additional,

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[5] After 1998, all operating LLW facilities were located in NRC Agreement States and no longer reported annual dose numbers to the NRC.

unnecessary meter of drift that needs to be excavated and developed results in addition, unnecessary radiological risk to workers due to external and internal exposure from natural radioactivity and external exposure due to man-made radioactivity once emplacement of waste packages begins.

DOE envisions subsurface construction activities, including drift excavation and development, to occur over the initial 5 year construction phase and extending into the first 22 years of the operations phase of the repository.

Collective dose to workers is calculated on a unit (per meter) basis for drift excavation by summing collective doses over the construction phase and the first 22 years of the operations phase for involved subsurface craft workers and then dividing this dose by DOE's estimated total drift length (67,915 m) as discussed further in Appendix D. Involved subsurface collective dose for the construction period is estimated by DOE to be 33 person-rem (DOE 2008a,, Table 4-23, Vol. I, Ch. 4, pg. 4-66).

Total collective dose to involved workers during the operations phase is estimated by DOE to be 4,200 person-rem. The collective dose to involved subsurface craft workers is calculated by multiplying this total by the ratio of involved subsurface craft FTEs during operations (4339) to total involved staff FTEs during operations (23,399) (DOE, 2008b, non-rad H&S folder; filename: CAlcPkg_HS1_Attch1_JLS_09-04-07.xls). The resulting collective dose to involved subsurface craft workers during operations is 779 person-rem.

Based on the above calculations, EPRI estimates that the total dose to involved subsurface craft works would be 812 person-rem. The resulting dose on a per meter basis is then 0.012 person-rem/m or 0.067 person-rem/waste package (assuming 5.6 m per average waste package). As discussed in Appendix A.3, a total of 7400 CSNF waste packages would be disposed of using DOE's assumed 21P/44B TAD for CSNF. Under EPRI Case 1, a total of 5,080 larger capacity TADs would be disposed – 2,320 less than assumed in the YMSEIS. This results in a reduction in worker dose associated with subsurface operations of 155 person-rem. Under EPRI Case 2, a total of 5,929 larger capacity TAD waste packages would be disposed – a 2,471 reduction in waste packages. This results in a reduction of worker dose associated with subsurface operations of 166 person rem.

B.3.5 Radiological Impacts Associated with Drip Shield Installation

The YMSEIS assumes that the annual individual dose associated with installation of the drip shields is 9.75 mrem per year, with a staffing of 10 persons per year, resulting in a total dose of 97.5 person-mrem per year. The repository closure phase is assumed to last for 10 years, although it is not clear from the YMSEIS whether the drip shield installation operations will take place during the entire 10-year operations-closure phase. If drip shield installation takes five years, the total dose would be 487.5 person-mrem. If it takes ten years, the total dose for drip shield installation would be 975 person-mrem. (BSC 2007)

B.4 Radiological Impacts Associated with A One-Year Delay of CSNF Shipment to Yucca Mountain

While the YMSEIS and its associated calculational package did not calculate the additional radiological risk associated with additional construction at reactor site Independent Spent Fuel

Storage Installations (ISFSIs), documentation associated with DOE's No Action Alternative did evaluate these impacts. If additional ISFSI construction is required while there is already CSNF in dry storage, DOE's No Action Alternative assumed that there would be an additional 170 person-mrem per additional cask loaded (Jason 1999, Rollins 1998). In addition, as noted in Table A-4 above, ISFSI operation and maintenance was estimated to incur additional worker radiation exposure of:

- 120 person-mrem per year per site surveillance
- 1,500 person-mrem per year per site for annual maintenance

The YMSEIS assumes that there are 75 commercial reactor sites. If nuclear operating companies are discharging 2,200 MTU of CSNF from plants on an annual basis, this would require between 160 and 220 dry storage systems per year for additional on-site storage (assuming between 10 and 14 MTU per system). Thus, assuming that each reactor site would have an operational ISFSI by the 2020 time period, this results in the following industry wide impacts:

- 9 person-rem per year for ISFSI surveillance activities
- 112.5 person-rem per year for ISFSI annual maintenance
- 27 to 37 person-rem per year for additional ISFSI construction (170 person-mrem per cask loaded).

B.5 References

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Jason 1999. Health and Safety Impacts during Controlled Long Term Storage of SNF and HLW in the U.S., Jason Technologies, April 1999, LSN #: DN2001094424.

Rollins 1998. Radiological Impacts for Scenario 1 at Commercial Nuclear Power Plants, Rollins, Tetra Tech NUS, October 1998, LSN #: DN2001483535

C

NON-RADIOLOGICAL WORKER IMPACTS

C.1 Non-Radiological Impacts at Reactor Sites

DOE considers non-radiological or industrial safety impacts to industry workers associated with CSNF storage at reactor sites and transport to Yucca Mountain, applying Bureau of Labor Statistics occupational hazard data from 2005 for total reportable cases (TRC) per 100 employees, lost workday cases (LWC) per 100 employees, and fatalities per 100,000 employees. Accordingly, EPRI utilized 2005 BLS data for consistency (BLS 2006a,b) Table C-1 provides data for relevant occupations.

Table C - 1
Occupational Injury and Fatality Rate Data for Relevant Occupational Categories in 2005 (BLS 2006a,b)

Category	NAICS code	TRC(per 100 FTE)	LWC (per 100 FTE)	Fatalities (per 100,000 FTE)
Construction	23	6.3	3.4	11.0
Warehousing and storage	493	8.2	5.4	17.6[a]
Truck transportation	483	6.1	3.9	17.6[a]
Rail transportation	482	6.0	4.5	17.6[a]
Utilities	22	4.6	2.4	3.6
Mining	21	3.6	2.2	25.6

[a]Fatalities for transportation and warehousing category, NAICS code 48-49

Fatality rates for specific high risk occupations that are relevant for Yucca Mountain construction and operation are also presented for illustration purposes in Table C-2..

Table C - 2
Selected Occupations with High Fatality Rates for 2005 (BLS 2006a,b)

	Fatalities (per 100,000)
Structural iron and steel workers	55.6
Driver/sales workers and truck drivers	29.1
Construction laborers	22.7

C.1.1 Non-Radiological Impacts Associated with Cask Loading and Handling

Loading and handling of canisters and cask systems represent is one of the key contributors to worker risk and is subject to substantial changes based on how and when DOE operates a repository. For the purposes of this report, EPRI estimates occupational impacts from canister/cask loading and handling operations in two ways. The first approach assumes a representative output of canisters on an annual basis and yields calculated hazards based on estimated person-hours required for that activity. The second approach evaluates the differential impact of DOE's decision to adopt a standardized TAD canister for CSNF with a 21 PWR/44 BWR fuel element capacity.

For the first approach, EPRI assumes that loading and handling of one canister/cask system requires 400 person-hours (0.20 FTE). This assumption is based on estimates for loading, on site transport, and emplacement of a TN-32 horizontal cask system (Dominion, 2002). Applying Bureau of Labor Statistics data for warehousing and storage occupations, this translates into the following non-radiological impacts for each canister/cask loaded at reactor sites:

- TRC = 0.016
- LWC = 0.011
- Fatalities = 0.000035

For the second approach, EPRI adopted the approach used by DOE in its Final SEIS (DOE 2008a). According to the YMSEIS, the analysis of industrial safety impacts was based on an average loading duration of 2.3 days per rail cask for PWR SNF and 2.5 days per rail cask for BWR SNF. DOE's analysis assumed truck cask loading times of 1.3 days per cask for PWR SNF and 1.4 days per cask for BWR SNF. (DOE 2008a, Section G.1.3). A total of 1,347 worker-years would be spent on loading activities for involved workers. DOE also calculated non-involved worker impacts, assuming that the non-involved workforce would be 25% of the involved workforce.

According the YMSEIS, DOE based incidence and fatality rates on Bureau of Labor Statistics (BLS) data for 2005 (BLS 2006a,b), referencing the data for workers in the transportation and warehousing industries to estimate impacts associated with loading SNF casks. The following assumptions were used to calculate worker impacts associated with loading TAD canisters with CSNF:

- 8.2 TRC per 100 FTE for Involved Workers (warehousing and storage, 2005)
- 5.4 LWC per 100 FTE for Involved Workers (warehousing and storage, 2005)
- 17.6 Fatalities per 100,000 workers for Involved Workers (transportation and warehousing, 2005) (DOE 2008a, Table G-3)

Utilizing the above assumptions, DOE calculated industrial safety impacts to involved workers as summarized in Table C-3. Impacts included 110 total recordable cases (TRC); 73 lost workday cases (LWC) and 0.24 industrial fatalities for loading activities for CSNF, DOE SNF, DOE HLW, and Naval SNF. Assuming changes in the number of CSNF packages loaded, consistent with ERPI Case 1 and EPRI Case 2, EPRI recalculated the industrial safety impacts in order to quantify the increase in the impacts associated with DOE's selection of a 21P/44B TAD rather than a higher capacity TAD design similar in capacity to DPCs being loaded at reactor sites.

Table C - 3
Estimated Industrial Safety Impacts to Involved Workers During Loading Operations

Impact Type	Impact		
	DOE YMSEIS	EPRI Case 1	EPRI Case 2
Total recordable cases	110	91	79
Lost workday cases	73	60	52
Industrial Fatalities	0.24	0.20	0.17

As shown in Table C-3, DOE's selection of a 21P/44B TAD rather than a higher capacity TAD design similar in capacity to DPCs being loaded at reactor sites today results in the following increased health and safety impacts to involved workers:

- 19 TRC
- 13 LWC
- 0.04 industrial fatalities

DOE's selection of a 21P/44B TAD rather than a higher capacity TAD and its assumption that seven commercial nuclear power plant sites would ship CSNF to Yucca Mountain using truck casks rather than DPCs or large capacity TADs results in 4,217 additional packages being loaded at reactor sites. This results in the following increased health and safety impacts to involved workers:

- 31 TRC
- 21 LWC
- 0.07 industrial fatalities

C.1.2 Non-Radiological Impacts Associated with ISFSI Operation and Maintenance

As part of its occupational health and safety calculations, DOE in its YMSEIS used the following assumptions for ISFSI operation and maintenance.

- Total inspection/security surveillance: 30 person-hours per year (0.015 FTE)
- Total maintenance: 30 person-hours per year (0.015 FTE)
- Total for ISFSI operational and maintenance: 60 person-hours per year (0.030 FTE)

Using the Bureau of Labor Statistics injury and fatality rates for the utility occupational category (NAICS code 22) yields the following projected annual impacts at each ISFSI site for surveillance/inspection:

- TRC 0.00069
- LWC 0.00036
- Fatalities 5.4×10^{-7}

Likewise, annual impacts at each ISFSI for routine maintenance are calcualated to be:

- TRC 0.00069
- LWC 0.00036
- Fatalities 5.4×10^{-7}

C.1.3 Non-Radiological Impacts Associated with ISFSI Expansion and Construction

Using BLS injury and fatality data for the construction industry and assuming that the estimated time associated with construction of one horizontal storage module is 1500 person-hrs (0.75 FTE) (Rollins, 1998), EPRI estimates the following non-radiological impacts associated with ISFSI expansion and construction of additional storage modules:

- TRC 0.047
- LWC 0.026
- Fatalities 0.000083

Assuming that the estimated time asociated with construction of an additional ISFSI storage pad is 7090 person-hrs (3.5 FTE) (Dominion, 2002), EPRI estimates the following non-radiological impacts associated with ISFSI expansion and construction of one additional ISFSI storage pad:

- TRC 22
- LWC 12
- Fatalities 0.00039

C.2 Non-Radiological Impacts to Workers During Transport

The YMSEIS idenfies the probablity of a rail transport accident to be 1.15 × 10-8 fatality/railcar-km (DIRS 178016-DOT 2005, all). For shipments involving 3 spent nuclear fuel casks (8 railcars total), the fatality rate was estimated to be9.20 × 10-8 accidents/train-km.

In the YMSEIS, the non-radiological fatality rate associated with rail accidents was estimated to be 1.15×10^{-8} fatality/railcar-km. For shipments involving three CSNF casks (8 railcars total), the fatality rate was estimated to be 9.20×10^{-8} accidents/train-km. Thus, a reduction in the number of cask shipments that results in a reduction in the number of train shipments would reduce the risk of transportation accidents and fatalities. (Source: DOE 2008b, p. 53)

The YMSIES identifes the probablity of truck transport accidents. Truck accident and fatality rates are state specific; however the average accident rates for trucks are:

- 5.34E-07 accidents per truck km

- 55E-08 fatalities per truck km

(Source: DOE 2008b, Attachment 8A Database)

C.3 Non-Radiological Impacts to Workers at Yucca Mountain

The YMSEIS estimated non-radiological health and safety impacts to workers at Yucca Mountain from industrial hazards using the Computerized Accident/Incident Reporting System (CAIRS) database. CAIRS is a DOE database that collects reports of injuries, illnesses, and accidents that occur at DOE sites. It records TRC and "days away, restricted or on job transfers", which is equivalent to the BLS LWC category. Table C-4 presents the non-radiological health and safety statistics used in the SEIS to calculate impacts to involved workers.

Table C - 4
DOE Occupational Injury and Fatality Data for Construction and Operations Periods from CAIRS Database

Project period	TRC	LWC	Fatalities	Source
Construction	2.0	0.86	0.55	DOE 2008a, Table 4-16), Section 4.1.7.1
Operations	1.4	0.58	0.55	DOE 2008a, Table 4-20, Section 4.1.7.1.2

The YMSEIS calculated the impacts to involved workers during construction, operation, monitoring and closure of the repository, as summarized in Table C-5. While the calculational packages that support the YMSEIS does contain a breakout of worker hours for each of the operational periods identified in Table C-5, EPRI was not able to identify the specific worker hours associated with handling of the TAD packages for receipt, waste package closure, aging and emplacement. Therefore, EPRI was not able to identify the increase in worker hours associated with DOE's decision to utilize a 21P/44B TAD package and rather and higher capacity TAD packages as described by EPRI Case 1. Similarly, EPRI was not able to identify the increase in worker hours associated with DOE's decision to utilize a 21P/44B TAD design and to ship CSNF using truck casks from seven commercial nuclear power plant sites.

EPRI has not quantified the additional industrial hazards to workers associated with the receipt and handling the 9,456 CSNF casks assumed in the DOE YMSEIS, rather than a total of 7,548 casks under EPRI Case 1 – a 20% reduction in the number of packages handled and emplaced. Similar, under EPRI Case 2, industrial hazards associated with handling 5,239 casks under EPRI Case 2 – more than a 40% reduction in packages handled – would be lower than the impacts associated with handling 9,456 CSNF casks as assumed by DOE.

Table C - 5
Impacts to Involved Workers During Construction, Operations, Monitoring and Closure Periods for a Yucca Mountain Repository

Impact Category/Operations Period	**Impact**
Construction	
▪ TRC	120
▪ LWD	50
▪ Fatalities	0.032
Operations – Surface Construction	
▪ TRC	53
▪ LWC	23
▪ Fatalities	0.015
Operations – Subsurface Construction	
▪ TRC	87
▪ LWC	37
▪ Fatalities	0.024
Operations – Emplacement Operations	
▪ TRC	160
▪ LWC	67
▪ Fatalities	0.064
Operations – Maintenance	
▪ TRC	68
▪ LWD	28
▪ Fatalities	0.027
Monitoring	
▪ TRC	320
▪ LWC	130
▪ Fatalities	0.31
Closure	
▪ TRC	320
▪ LWC	150
▪ Fatalities	0.15
Source: DOE 2008b, H&Snonrad File, Attachment 1	

C.3.1 Non-Radiological Impacts Associated with Receipt, Handling, and Aging of CSNF

Not estimated as a separate category. Refer to Section C.3 above.

C.3.2 Non-Radiological Impacts Associated with Unloading Additional Dual-Purpose Canisters

Not estimated as a separate category. Refer to Section C.3 above.

C.3.3 Non-Radiological Impacts Associated with Management of Empty DPCs as Low-Level Radioactive Waste

In terms of non-radiological hazards, the handling of empty DPCs will also incur non-trivial risks to workers due to the routine hazards of handling heavy materials. Each empty DPC can weigh on the order of 36,000 lbs to 58,000 lbs. For Yucca Mountain work, DOE uses occupational hazard figures derived from its own experience as documented agency's CAIRS database. For the operational phase, these occupation risk numbers are: 1.4 TRC per 100 FTEs, 0.58 LWC per 100 FTEs, and 0.55 fatalities per 100,000 FTEs. Disposal of the empty DPCs offsite would, likewise, impose non-radiological risks to workers at commercial facilities. For these workers, it would be appropriate to apply BLS data (from Section B.1):

- 8.2 total recordable cases (TRC) per 100 FTE for Involved Workers (warehousing and storage, 2005)
- 5.4 lost workday cases (LWC) per 100 FTE for Involved Workers (warehousing and storage, 2005)
- 17.6 Fatalities per 100,000 workers for Involved Workers (transportation and warehousing, 2005) (DOE 2008a, Table G-3)

The current DOE proposed approach does not call for unloading of DPCs at generator sites; however, it in conceivable that such a burden could be shifted to utilities and other ISFSI operators. Any LLRW management activities resulting from the unloading of DPCs at the plant site would present occupational risk to those involved workers.

C.3.4 Non-Radiological Impacts Associated with Additional Subsurface Construction Resulting from the Exclusive Use of Low Capacity TAD Canisters for CSNF

DOE's decision to use the 21P/44B TAD canisters rather than higher capacity TADs will require the excavation of more emplacement drifts and associated access drifts than if higher capacity TADs and/or DPCs were accommodated in the proposed action. Accordingly, each additional, unnecessary meter of drift that needs to be excavated and developed results in addition, unnecessary radiological risk to workers due to external and internal exposure from natural radioactivity and external exposure due to man-made radioactivity once emplacement of waste packages begins.

DOE proposes subsurface construction activities, including drift excavation and development, occurring over the initial 5 year construction phase and extending into the first 22 years of the operations phase of the repository.

EPRI calculated the non-radiological occupational risks associated with drift excavation on a unit (per meter) basis by summing the respective occupational health and safety categories (TRC, LWC, and fatalities) over the construction phase and operations phase for involved subsurface craft workers and then dividing by DOE's estimated total drift length (67,915 m). Occupational health and safety numbers for subsurface construction during the construction phase are calculated by applying the ratio of the subsurface craft FTEs (336) to the total FTEs (5,886) for the period (DOE 2008b, non-rad H&S folder; filename: CAlcPkg_HS1_Attch1_JLS_09-04-07.xls). As shown in Table C-6, EPRI calculated that the fraction of FTE associated with subsurface craft workers is 0.057. Table C-7 summarizes the worker health and safety impacts during the construction phase from the YMSEIS – with 117.2 TRC, 50.2 LWC, and 0.032 fatalities. Using the subsurface craft worker fraction calculated in Table C-6, EPRI estimated the worker impacts during the construction phase for subsurface workers – 6.69 TRC, 2.86 LWC, and 0.0018 fatalities. Occupational health and safety numbers are explicitly reported for subsurface construction during the operations phase, as shown in Table C-7. During the operations phase, subsurface construction results in occupational health and safety impacts of 87.08 TRC, 37.29 LWC, and 0.024 fatalities.

Table C - 6
FTE During Construction Phase (2012 – 2016)

	FTEs
Subsurface Craft FTE	335.75
Total FTE	5886
Subsurface Craft Fraction	0.057

Table C - 7
Estimated Worker Health and Safety Impacts During Construction and Operation

Construction Phase – Total Impacts	**Cases**
TRC	117.2
LWC	50.2
Fatalities	0.032
Construction phase – subsurface construction only (calculated)	**Cases**
TRC	6.69
LWC	2.86
Fatalities	0.0018
Operations phase - subsurface construction	Cases
TRC	87.08
LWC	37.29
Fatalities	0.024

Summing the non-radiological impacts associated with construction of subsurface facilities during the construction and operations phases, EPRI calculated impacts of 97.77 TRC, 40.14 LWC, and 0.026 fatalities as shown in Table C-8. Assuming that the total excavated drift length in the repository is 67,915 meters, EPRI calculated the number of worker impact cases per meter as shown in Table C-8. Assuming that each waste package occupies a drift length of approximately 5.6 meters as discussed previously in Appendix B.7, EPRI estimates the number of worker impact cases per waste package emplaced.

Table C - 8
Unit Non-Radiological Occupational Risks Associated with Subsurface Construction

Worker Impacts	Cases	Cases per Emplacement Meter	Cases per Waste Package Emplaced
TRC	93.77	1.4×10^{-3}	7.7×10^{-3}
LWC	40.15	5.9×10^{-4}	3.3×10^{-3}
Fatalities	0.026	3.8×10^{-7}	2.1×10^{-6}

C.3.5 Non-Radiological Impacts Associated with Drip Shield Installation

The YMSEIS assumes a staffing level of 10 persons per year associated with drip shield installation. The repository closure phase is assumed to last for 10 years, although it is not clear from the YMSEIS whether the drip shield installation operations will take place during the entire 10-year operations-closure phase. On annual basis, then, non-radiological impacts to workers during drip shield installation are calculated as follows using DOE's industrial safety statistics for a 10 FTE workforce:

- TRC 0.82 per year
- LWC 0.54 per year
- Fatalities 0.0018 per year

Thus, over an assumed five year period for drip shield installation there would be 4.1 TRC, 2.7 LWC, and 0.009 fatalities. If drip shield installation takes place over a ten-year period, the estimated non-radiological worker impacts would be 8.2 TRC, 5.4 LWC, and 0.018 fatalities.

C.3.6. Non-Radiological Impacts Associated with Over-Design of Surface Facilities for Seismic Considerations

Due to the classification of facility details as "Official Use Only" resulting in their omission from the publicly available version of the License Application, EPRI attempted to evaluate the occupational consequences on a more generic, semi-quantitative level using a stylized approach based on the available dimensions for the WHF footprint, typical above-grade height, and wall thickness. EPRI assumed for the purpose of this illustration a WHF facility comprised solely of a rectangular concrete shell. As part of this approach, EPRI ignored the contributions from roof, base mat/pad, and interior walls. The data and assumptions are listed below:

- Dimensions from DOE , 2008 LA (DOE, 2008c; p. 1.2.5-3):
  - ITS footprint of waste handling facility = 385 ft. x 300 ft.
  - Typical height of facility above grade = 80 ft.
  - Exterior wall thickness = 4 ft.
- Conservative assumptions:
  - a simple four sided building shell with above dimensions
  - not considering contribution of internal walls (unknown)

  - not considering contribution of base mat/foundation/pool structures (assume to be appropriate)
  - not considering contribution of roof (unknown/assume to be appropriate)
  - not considering shrinkage of concrete upon drying
- Other Assumptions
  - neglecting volume consumed by rebar, openings (more than offset by conservative simplification of building)
  - capacity of a typical ready mixed concrete truck = 20 cu yd. or 540 cu. ft. (Clark et al., 2001)

Using these assumptions and data, the resulting volumes are calculated:

- Total wall volume = 438400 cu. ft. = 812 truck loads
- Volume reduction for 10% reduction in wall thickness = 43840 cu. ft. = 81 truck loads
- Volume reduction for 25% reduction in wall thickness = 109600 cu. ft. = 203 truck loads

Any unnecessary and unjustified conservatism in the construction of WHF and other surface pre-closure facilities result in incremental increases in worker risk due to well-documented occupational hazards. In addition to the often repeated fact that the construction industry is a perennial leader in occupational injury and fatality rates, the Bureau of Labor Statistics has also singled out three specific occupations that exhibited exceptionally high fatality rates in 2005: structural iron and steel workers, truck drivers, and construction laborers.

Table C - 9
Selected Occupations with High Fatality Rates for 2005 (BLS, 2006a)

	Fatalities (per 100,000)
Structural iron and steel workers	55.6
Driver/sales workers and truck drivers	29.1
Construction laborers	22.7

The reinforcement of concrete structures to withstand seismic loads directly involves the contribution of all three of these high-risk occupations for the preparation of appropriate concrete forms, assembly of additional rebar, and pouring of additional concrete. Additional concrete also results in additional truck deliveries that could number in the 100’s to 1000’s for the case of an over-designed facility. Accordingly, the purposeful over-design (beyond standard engineering margins) for seismic or any other hazard represents unnecessary and unjustified imposition of risk to the involved workers.

Table C - 10
Relevant BLS[a] and DOE[b] Non-radiological Injury and Fatality Rates

Category	TRC	LWC	Fatalities
BLS - construction	6.3	3.4	11.0
BLS - warehousing and storage	8.2	5.4	17.6[c]
BLS - truck transportation	6.1	3.9	17.6[c]
DOE - construction period[b]	2.0	0.86	0.55

[a]BLS, 2006a,b
[b]DOE 2008a, Table 4-16), Section 4.1.7.1
[c]Fatalities for transportation and warehousing category, NAICS code 48-49

It should be noted that DOE's injury and fatality rates are substantially lower that reported by BLS. DOE does not differentiate between specific trades and occupations such as iron workers.

In addition to occupational consequences, the over-design of facilities also consumes significant quantities of materials and resources that would have beneficial uses elsewhere, especially in terms of concrete (cement and aggregate) and rebar (iron/steel).

C.4 Non-Radiological Impacts Associated with a One-Year Delay of CSNF Shipment to Yucca Mountain

The major consequence of a delay in Yucca Mountain becoming operational is that existing inventories of CSNF will remain for a longer period of time at reactor sites and other commercial facilities and additional quantities of CSNF will need to be stored in both wet and dry storage. These burdens result in additional occupational health risk to workers at reactor storage sites (and other commercial facilities) associated with fuel, canister, and cask handling operations, onsite transport and emplacement operations, routine surveillance and maintenance activities, and construction of additional storage capacity. For this report, EPRI focused on the ISFSI related activities.

The YMSEIS assumes that there are 75 commercial reactor sites. Accordingly, EPRI estimates the industry wide non-radiological impacts of a one-year delay of CSNF shipments to Yucca Mountain by extrapolating the impacts described in Sections C.1.2 and C.1.3 of this Appendix for ISFSI operation and expansion, respectively, to the SEIS inventory of 75 reactor sites, assuming that each reactor site would have an operational ISFSI by the 2020 time period. These results are summarized in Table C-11.

Table C - 11
Industry Wide Non-Radiological Impacts Associated with a One-Year Delay

Activity	Annual Injuries and Fatalities (cases)
ISFSI Surveillance and inspection	0.052 TRC 0.027 LWC 4.1×10^{-5} fatalities
ISFSI Maintenance	0.052 TRC 0.027 LWC 4.1×10^{-5} fatalities
Additional storage module construction at existing ISFSI[a]	7.5 – 10 TRC 4.2 – 5.7 LWC 0.013 – 0.0189 Fatalities
ISFSI pad construction[b]	22 TRC 12 LWC 3.9×10^{-4} fatalities

[a]Based on TN-32 horizontal storage module (Rollins, 1998) and annual requirement of 160 – 200 dry storage systems for 75 commercial reactor sites.

Additionally, in the event that either existing ISFSI pad capacity at a particular site is full or does not exist, the construction of a new pad could become necessary. Table C-12 includes the non-recurring occupational consequences associated with the construction of one ISFSI pad from Section C.1.3.

Table C - 12
Non-Radiological Impacts Associated with a the Need to Construct One Additional ISFSI Pad

Activity	Total Injuries and Fatalities (cases)
ISFSI pad construction[b]	22 TRC 12 LWC 3.9×10^{-4} fatalities

[b]Based on 7090 person-hours estimate for construction of one ISFSI pad for storage of up to 28 TN-32 horizontal storage modules (Dominion, 2002).

C.5 References

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D
REPOSITORY SUBSURFACE EXCAVATION

The EPRI analysis presented in this report relies upon assumptions, estimates, and specifications pertaining to subsurface excavation and construction. For clarity, these are summarized below.

Projected Repository Subsurface Construction Requirements (DOE, 2008):

- Total drift length = 67,915 m
- Total drift length for emplacement of WPs = 65,209 m
- Drift diameter = 5.5 m
- Average/typical emplacement drift length = 600 m
- Approx. number of emplacement drifts = 108 in 4 panels
- Total volume of excavated rock = 6.5×10^6 m^3
- Volume of excavated rock for emplacement of WPs = 6.2×10^6 m^3
- Average length for 1 WP = 5.6 m
- Volume of excavated rock per meter of drift = 24 m^3
- Volume of excavated rock per average waste package = 133 m^3

Excavation volumes calculated assuming cylindrical geometry

- Total number of waste packages for emplacement (in TSPA) = 11,629
- Total number of TADs for emplacement = 7,400

Reference

DOE 2008. Yucca Mountain Repository License Application. U.S. Department of Energy, Office of Civilian Radioactive Waste Management, DOE/RW-0573, Rev. 0, June 2008.

E
MATERIALS AND IMPACTS ASSOCIATED WITH KEY REPOSITORY SYSTEMS

Overly conservative design and certain operational decisions will result in the consumption of materials and manufacture and shipping of additional heavy components to either utility sites or Yucca Mountain, incurring non-trivial risks to workers as well as the public.

Overdesign of Yucca Mountain surface and sub-surface facilities incurs an additional, unnecessary risk burden to workers for every additional cubic meter of concrete poured and each meter of rebar used. While EPRI does not calculate total additional risk associated with such conservatism in the repository design, it is clear that such risks are significant in that the construction industry is routinely cited as one of the most hazardous occupations by the Bureau of Labor Statistics.

There are two primary scenarios for which impacts from manufacturing and transportation of heavy components are pertinent:

- unnecessary use of titanium drip shields, and
- additional emplacement drift construction and the associated infrastructure required by the disposal of smaller waste packages (i.e., containing less CSNF than necessary).

E.1 Unnecessary Use of Titanium Drip Shields

By invoking the use of drip shields, the DOE is incurring substantial resource demands for titanium, a material of significant strategic importance and of limited domestic availability. DOE estimates that its projected schedule for drip shield manufacture will result in consumption of 22% of present day annual U.S. production of Ti for a limited period of time as shown in Table E-1. Moreover, manufacture of the drip shields incurs occupational risks to involved workers. The YMSEIS (DOE, 2008a) estimates that 11,500 drip shields will be used under the Proposed Action. And as a heavy component, the YMSEIS assumes that 25 drip shields will be shipped per rail car, with a total of 460 shipments. The YMSEIS assumed a shipping distance of 3,464 km, resulting in pollution health effect fatalities of 0.028 and vehicle fatalities of 0.036 – or *total fatalities of 0.064 associated with the transport of drip shields from manufacturing facilities to the proposed repository*. (DOE 2008b, Transportation File, Attachment 12, Other materials.

In addition to the fatalities associated with transport of the drip shields, offsite manufacturing of 11,500 drip shields is estimated to take3.5 million labor hours. The YMSEIS analysis of off-site manufacturing health and safety impacts assumed 9.1 injuries per 100 full-time worker years and 3.29 fatalities per 100,000 worker years. *This results in 159 injuries and 0.609 fatalities associated with off-site manufacturing of the drip shields.* (DOE 2008b, Offsite Manufacturing File, Attachment A.)

Table E - 1
Materials Required for Repository Construction and Component Manufacturing (DOE, 2008a Final SEIS, Tables 4-30, 4-36

Material	Quantity	Proj. percentage of U.S. annual production
Concrete	490,000 m^3	
Cement	190,000 metric tons	
Carbon Steel	280,000 metric tons	
Copper	670 metric tons	
Copper*	140	0.0004%
Titanium*	54,000 metric tons	22%
Chromium*	100,000 metric tons	1.8%
Nickel*	120,000 metric tons	3.6%
Molybdenum	27,000 metric tons	1.9%

*Quantities are for repository components only, not total repository construction.

E.2 Additional Infrastructure to Support Additional Waste Package Emplacement

Each additional (unnecessary) WP emplaced at YM would require 5.6 m of drift and associated infrastructure, including one emplacement pallet, DS segment, and one TAD canister with outer waste package. The total quantities of materials associated with repository construction and component manufacture are summarized in Table D-1. Table E-2 summarizes the total number of repository components manufactured offsite. The YMSEIS estimates total worker injuries of 1,686 and total worker fatalities of 0.61 associated with manufacture of offsite components under the health and safety impact assumptions identified in the note on Table E-2.

The YMSEIS analysis of off-site manufacturing health and safety impacts assumed 9.1 injuries per 100 full-time worker years and 3.29 fatalities per 100,000 worker years. Assuming that the manufacturing of off-site components takes a total of 37 million labor hours and an average worker year is 2000 hours, the YMSEIS calculated total worker years of 18,500, resulting in 1,685 injuries and 0.61 fatalities associated with off-site manufacturing. (DOE 2008b, Offsite Manufacturing File, CalcPkg_Manufacturing1_AttchA.xls).

E.3 References

DOE 2008a. Final Supplemental Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada, DOE/EIS-0250F-S1, LSN #: DEN001593669.

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Table E - 2
Repository System Components Manufactured Off-Site

Component	Materials	Number	Weight (Metric tons)	Number of Shipments
Waste Packages (outer)	Alloy 22	11,200	22-34	5,589
TAD Canisters	Stainless steel	7,400	29 - 31	3,700
Emplacement pallets	Alloy 22 and stainless steel	11,200	2	5,302
Titanium drip shields (section)	▪ Grade 7 Ti – surface plates ▪ Grade 29 Ti- structural components ▪ Alloy 22 - base	11,500	4.9	460
Aging overpacks (carbon steel components)	Carbon steel liner and shell	2,500	43	1,250
Note: The YMSEIS estimates health and safety impacts associated with off-site manufacturing of repository components to be 3.3 fatalities per 100,000 worker years; 9.2 illness/injuries per 100 FTE. A 24 year manufacturing period is assumed for all components except drip shields. Drip shields are manufactured over a 10 year period.				

Source: DOE 2008a, Section 4.1.14.2; DOE 2008b, Offsite Manufacturing File, Attachment A.

F

METHODOLOGY FOR EPRI'S INDEPENDENT PROBABILISTIC VOLCANIC HAZARD ANALYSIS

EPRI has recently conducted an independent assessment of the likelihood of a future volcanic event occurring at the proposed Yucca Mountain repository site. A more detailed report on this issue will be released later this year. The assessment methodology adopted in the EPRI study was based on same methodology applied in the 1996 Probabilistic Volcanic Hazard Analysis (PVHA) report (CRWMS M&O, 1996, pp. 2-19). The purpose of EPRI's study was to independently develop new insights and probability estimates for future volcanism based on the more recent, extensive geological and structural data obtain in the last 12 years in the Yucca Mountain region (YMR).

EPRI's PVHA study includes consideration of new geochemical, geophysical, seismological, geodetic and age-dating data collected since the 1996 PVHA report (e.g., Brocher et al., 1998; Day et al., 1998; Perry et al. 1998; Fridrich, 1999; Fridrich et al. 1999; Potter et al., 2002; 2004; Perry et al., 2005; Valentine et al., 2005; 2006; Parson et al., 2006; Valentine and Krough, 2006; Valentine and Perry, 2006; Gaffney et al., 2007; Perry, 2007; Valentine and Perry, 2007; Valentine et al. 2007; Keating et al, 2008). In particular, EPRI's calculation includes information from drilling (Perry et al., 2005; Perry, 2007) and characterization (i.e. age dating) of various anomalous features identified by recent high resolution aeromagnetic surveys (O'Leary et al., 2002; Perry et al., 2005) buried under alluvial deposits that have been speculated to be additional volcanic centers (Perry et al, 2004; Smith and Keenan, 2005). Furthermore, EPRI's independent update to the 1996 PVHA report includes consideration of structural factors that demonstrably have controlled the actual eruptive location of volcanic centers that have occurred in the Yucca Mountain region in the last 12 million years (Valentine and Perry, 2006; 2007; Gaffney et al., 2007; Keating et al, 2007). As noted by the NRC's Advisory Committee and Nuclear Waste (ACNW) report on volcanism (ACNW, 2007, pp. 63), for example, there has been no igneous intrusion into Yucca Mountain block in the last 10 million years.

The approach taken by EPRI follows that used in the 1996 PVHA (CRWMS M&O, 1996). The approach involves defining an igneous (volcanic) event that may intersect the footprint of the proposed repository within the next 10,000 to 1,000,000 years. The calculation requires that an igneous event be well defined and its characteristic features be quantified, and the identification of factors that govern the location and timing of a possible future igneous event in the YMR. By following a similar approach as the 1996 PVHA calculation, results from EPRI's calculation may be compared and evaluated to results in the 1996 PVHA (CRWMS M&O, 1996) and a planned PVHA-U (the updated version of the 1996 PVHA) by the DOE. The estimated annual frequency of intersection in the 1996 PVHA (CRWMS M&O, 1996) is expressed as:

$$\nu = \frac{N(R,T)}{T} \cdot \frac{a_r}{A_R}$$

where, N(R,T) is the number of events that have occurred in region R in time period T, A_R is the area of region R and a_r is the area of the repository. The above equation is expanded to the following expression to account for alternative temporal and spatial models (CRWMS M&O, 1996):

$$\nu_I = \iint_R \int\int \lambda(x,y,z)\, P_I(x,y)\, dx\, dy = \frac{N(R,T)}{TA_r} \iint_r dx\, dy = \frac{N(R,T)}{T} \cdot \frac{a_r}{A_R}$$

where, λ(x,y,t) is the rate density function (frequency of events per unit time per unit area), and P_I is the conditional probability (for a point source event, P_I = 1 inside the effective region of interest *r*, and 0 everywhere else). λ(x,y,t) is separated into two parameters: *λ(t),* rate parameter (*N(R,T)/T*), and *f(x,y)* spatial density ($1/A_R$). The probability calculation requires an understanding of an expected igneous event in the area of interest as well as an assessment of the spatial and temporal parameters.

The framework for EPRI's probability calculation is divided into fours steps. The first step is a review of recent data and development of EPRI's independent *conceptual model* for an expected igneous event in YMR in the next 10,000 to 1,000,000 years. The second step *defines EPRI's expected igneous event* that may intersect the repository including its characteristic features. The third step identifies EPRI's *region of interest* and factors that influence the spatial occurrence of an expected igneous event using a logic tree to illustrate alternative spatial as well as temporal models. The fourth step identifies and discusses the *time* of interest and *duration* of events.

For its Step 1 development of an independent conceptual model, EPRI evaluated trends in Yucca Mountain field data that includes geochemistry, volume, and location of volcanoes in YMR, as well as recent tectonic models, EPRI believes that if an eruption were to occur in YMR in the next 10,000 to 1,000,000 years, it would occur within the Crater Flat area, along a pre-existing fracture oriented perpendicular (N30E) to the least compressive stress field of the region and with a dip angle approaching vertical. The volcanic material would be alkali basalt, with eruption characteristics similar to volcanoes located within the Crater Flat area typified by the Lathrop Wells volcano. Furthermore, extensional trends in the YMR indicate the NE part of the basin (i.e., the location of the repository) will be less prone to future eruptions than the SW region.

"Event definition" in EPRI's Step 2 describes the expected ranges in characteristics of an igneous event that could intersect the repository at its proposed depth of 200-300 m below the surface of Yucca Mountain. At repository depths, the intrusion of igneous material occurs as a sheet-like dike; if this dike reaches the surface, the initial linear fissure eruption rapidly evolves into a eruptive conduit that can lead to formation of a scoria cone. Therefore, EPRI considers only dikes in its event definition; sills and conduits are considered to be features that develop after a dike has reached the surface. Important dike characteristics in the EPRI event definition include dike length and dike azimuth.

The region of interest (Step 3) in EPRI's PVHA analysis is defined by two areas, one large area and one smaller region. The larger region encompasses areas around the Yucca Mountain block in which the repository is located, to include Jackass Flats to the east, areas north such as Thirsty Mesa and Sleeping Buttes volcanoes, and areas south into the Amargosa Valley, and areas west bounded by the Bare Mountain fault. The smaller region considered by EPRI is essentially the Crater Flat structural domain with boundaries defined by faults: the Bare Mountain fault to the west, the Yucca Mountain fault to the north and the Gravity fault to the east. The larger region is used to evaluate each volcanic event in YMR with respect to event definition and its relevance on the spatial and temporal models for predicting a future igneous event. The smaller region defines EPRI's area of interest for its spatial model.

Two spatial models are considered in the EPRI analysis; a Fault Capture Model, and a No Fault Capture Model. The Fault Capture Model is based on recent DOE studies that demonstrate how low volume (< 1.0 km^3) magmas tend to ascend through the crust along the path of least resistance (Valentine and Perry, 2006; 2007; Gaffney et al., 2007; Keating et al, 2008). Initially magma will migrate through the lithosphere as a self-propagating dike following a direction (N30E in the YMR) that is perpendicular to the regional least compressive stress direction. As the dike approaches the surface, it will intersect and follow a fracture with a similar azimuth (N30E) and a steep dip angle (> 60^o). In EPRI's Fault Capture Model, only pre-existing faults are considered as probable locations for dikes and relative probabilities are assigned to faults that have been mapped in the Yucca Mountain region (Day et al., 1996; Potter et al., 2002; 2004; Perry, 2007) based on fault azimuth relative to the regional stress field (Stock et al., 1985). As an alternative, EPRI also considers a No Fault Capture Model in which it is assumed magma will ascend in a self-propagating dike that will reach with little influence from the pre-existing structure or topography. The dike will follow a path that is perpendicular to the least compressive stress direction. Probability distribution for event azimuth is assigned with respect to the regional stress field. This alternative model accounts for the uncertainty of an event that may not follow the Fault Capture Model. Both models consider lithostatic pressure and cumulative extension data in their evaluation of the location of a future event.

Finally, in Step 4 EPRI also considers temporal relationships and patterns of past eruptions as models for possible future eruptions in the YMR. In brief, EPRI evaluates two temporal conceptual models, one referred to as the Spatial Cluster Model and the other the Fault Initiated Cluster Model. The Spatial Cluster Model assumes that events are controlled by a regional tectonic event that initiates partial melting in the lithospheric mantle in one of the structural domains with the YMR. The Fault-Initiated Cluster Model assumes expected events are associated with localized fault movement.

Based on the more recent geological and structural data obtained by the US DOE (i.e., Valentine et al., 2005; 2006; Parson et al., 2006; Valentine and Krough, 2006; Valentine and Perry, 2006; Gaffney et al., 2007; Perry, 2007; Valentine and Perry, 2007; Valentine et al. 2007; Keating et al, 2008) and its own independent spatial and temporal models for controlling factors for the occurrence and eruption of igneous (volcanic) events, EPRI calculated a time-dependent probability of a future event intersecting a repository at Yucca Mountain (see Figure E-1). For a time 10,000 years after repository closure, EPRI's estimated range for igneous-event probability is 0.0 to 1.3 x 10^{-8} per year, with a mean value of 3.7 x 10^{-9} per year. For a period 1,000,000 years after repository closure, the estimated range for igneous-event probability is 0.0 to 7.3 x 10^{-9} per year, with a mean value of 3.0 x 10^{-9} per year. The decrease in probability values between 10,000 and 1,000,000 years (Figure F-1) is attributable to the time-dependent influence of EPRI's Spatial Cluster Model (i.e., events triggered by regional tectonic episode) imposed on the baseline of the Fault-Induced Cluster Model.

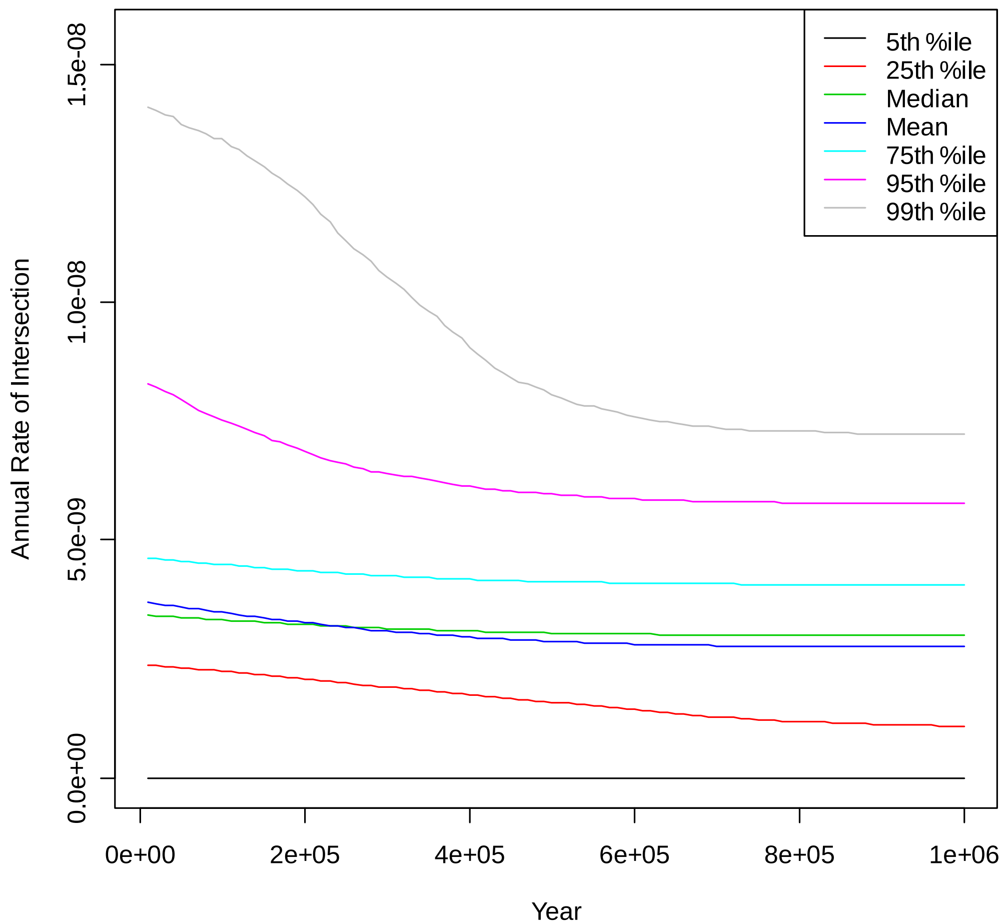

Figure F - 1
Calculated Probability for a Future Igneous Event Intersecting a Repository Located at Yucca Mountain, Nevada

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G

POTENTIAL ECONOMIC IMPACTS

In additional to the radiological and non-radiological impacts associated with DOE's decision to utilize the 21P/44B TAD canister rather than higher capacity canisters, there will also be economic impacts for nuclear operating companies. These economic impact include:

- Increased costs associated with loading additional packages at reactor sites
- Increased costs associated with transporting additional CSNF casks
- Increased costs to the DOE program associated with
  - The purchase of additional TAD canisters for transport, aging, and disposal
  - The purchase of additional waste packages for CSNF

EPRI has estimated the increased costs associated with DOE's decision to utilize the 21P/44B TAD canister rather than higher capacity canisters, as discussed in the sections below.

Under EPRI Case 1 assumptions, cost savings associated with using higher capacity TADs were estimated to be:

▪ At reactor loading costs	\$0.38 billion
▪ Transport costs	\$0.33 billion
▪ Disposal costs	\$3.14 billion
▪ Total potential savings:	\$3.85 billion

Unde rEPRI Case 2 assumptions, cost savings associated with using higher capacity TADs and assuming a minimal amount of CSNF is shipped by truck were estimated to be:

▪ At reactor loading costs	\$0.44 billion
▪ Transport costs	\$0.41 billion
▪ Disposal costs	\$3.33 billion
▪ Total potential savings	\$4.18 billion

G.1 Increased Cost Associated With Cask Loading and Handling At Reactor Sites

In calculating the costs associated with loading CSNF at reactor sites, EPRI assumed that TADs and DPCs would have a loading cost of \$200,000 per package. Truck casks were assumed to have a loading cost of \$50,000 per package. Using the number of packages estimated by EPRI in Appendix A, EPRI estimates that under EPRI Case 1, loading costs at reactor sites could be reduced by \$0.38 billion if DOE adopted larger capacity TAD packages rather than the 21P/44B TAD design as shown in Table G-1. Under EPRI Case 2, loading cots at reactor sites could be reduced by \$0.44 billion if DOE adopted a larger capacity TAD package and truck sites idenfied by DOE in the YMSEIS instead shipped CSNF in large capacity TADs.

DOE's YMSEIS assumes that all TADs loaded with CSNF at reactor sites will be stored at reactor ISFSIs prior to being transported to the repository for disposal. Thus, in addition to the increased costs associated with loading a greater number of 21P/44B TAD canisters, there will be an increase in the size of the ISFSI storage pad needed to store the additional TAD packages at reactor sites, compared to storing a smaller numer of higher capacity TADs or DPCs. EPRI has not attemped to quanitify the incremental ISFSI pad construction costs associated storing additional 21P/44B TAD packages at reactor sites since these costs would be site specific.

Table G - 1
Estimated Costs Associated with Cask Loading and Handling At Reactor Sites

Package Type	Loading Cost/Package	DOE YMSEIS	EPRI Case 1	EPRI Case 2
TAD 21P/44B	$200,000	$1.3 billion		
Large Capacity TAD 24P/32P,61B,68B	$200,000		$0.92billion	$0.99 billion
DPC	$200,000	$0.06 billion	$0.06 billion	$0.06 billion
Truck	$50,000	$0.13 billion	$0.13 billion	0
Total Cost		$1.49 billion	$1.11 billion	$1.05 billion
Cost Reduction			$0.38 billion	$0.44 billion

G.2 Increased Costs Associated With Transporting CSNF

The YMSEIS calculated a 2,833 total rail shipments (assuming three casks per train) for CSNF, DOE and Navy SNF, and HLW; and 2,650 truck shipments. As summarized in Table G-2, under EPRI Case 1 there would be an estimated 2,074 rail shipments, assuming three casks per train, and 2,650 truck shipments. Under EPRI Case 2 there would be an estimated 2,186 rail shipments (assuming 3 casks per train) and 2 truck shipments. The estimated rail shipments in the DOE YMSEIS, EPRI Case 1 and EPRI Case 2 include shipments of CSNF, DOE and Navy SNF, and DOE HLW.

DOE's July 2008, "*Analysis of the Total System Life Cycle Cost (TSLCC) of the Civilian Radioactive Waste*" (DOE 2008c) assumes that the costs for transport operations execution will be $3.12 billion to transport a total of 4,239 truck casks and 16,619 rail casks containing CSNF, DOE HLW and DOE SNF. EPRI estimated the unit costs per cask transported using data from DOE's 2008 TSLCC. EPRI assumed that the cost to transport one truck cask from reactor sites to Yucca Mountain would be $50,000. Thus, using DOE's data from the 2008 TSLCC, truck cask transportation would account for $211.95 million out of the total $3.12 billion. Dividing the remaining $2.91 billion by 16,619 rail casks assumed in the 2008 TSLCC, results in a cost per rail cask shipment of $175,100 per cask. It should be noted that the number of shipments in the 2008 TSLCC is higher than those considered by EPRI in this report since the 2008 TSLCC is based on total CSNF arisings of 109,300 MTU as well as all of the DOE SNF and HLW, and Navy SNF. EPRI's analysis considers the quantities of CSNF considered under the Proposed Action for a 70,000 MTU repository.

As shown in Table B-5, under the assumptions in the YMSEIS, the cost to ship CSNF would be approximately $1.324 billion. If rail shipments of CSNF utilized higher capacity casks than the 21P/44B TAD design as assumed in EPRI Case 1, the estimated cost to transport CSNF would be $990 million, a reduction of $334 million compared to cost for shipment of CSNF using the 21P/44B TAD. If rail shipments of CSNF utilized higher capacity TADs and the truck sites identified in the YMSEIS instead shipped by higher capacity TADs, the estimated cost to transport CSNF would be $917 million, a reduction of $407 million compared to cost for shipment of CSNF using the 21P/44B TAD and truck casks.

Table G - 2
Estimated Costs Associated with Transport of CSNF to Yucca Mountain

Scenario	Number of Casks	Estimated Transport Cost (Millions $)
DOE YMSEIS (70,000 MTU)		
▪ **Rail Casks Shipped**	6,806	$1,192
▪ **Truck Casks Shipped**	2,650	$132
Total Transport Cost		$1,324
EPRI Case 1		
▪ **Rail Casks Shipped**	4,898	$858
▪ **Truck Shipped**	2,650	$132
Total Transport Cost		$990
EPRI Case 2		
▪ **Rail Casks Shipped**	5,235	$917
▪ **Truck Shipped**	4	$0.2
Total Transport Cost		$917

G.3 Increased Costs To Handle and Disposal of CSNF

The YMSEIS assumed that a total of 7,400 TADs would be used for CSNF disposal under the proposed action (DOE 2008a, Table 4-32). As noted in Appendix A, the YMSEIS assumes that a total of 6,499 TADs are loaded with CSNF at reactor sites, leaving a total of 901 TADs to be loaded with commercial SNF that is shipped in the 307 DPCs and 2,650 truck casks. Under EPRI Case 1, a total of 4,591 higher capacity TADs are assumed to be loaded at nuclear power plant sites. If the CSNF shipped to the repository in DPCs and truck casks are repackaged at the repository into higher capacity TAD packages (32P, 68B), EPRI estimates that 489 packages would need to be loaded at the repository. Under EPRI Case 2, a total of 4,928 higher capacity TADs are assumed to be loaded at reactor sites. Under this scenario, there were two truck casks containing CSNF, which is assumed to be transferred to 1 TAD canister at the repository.

The 2008 TSLCC assumes that a PWR TAD will cost $700,000 and a BWR TAD will cost $800,000. For simplification, EPRI assumed an average TAD cost of $750,000. Under the YMSEIS assumptions, EPRI estimates that the cost of TAD canisters to dispose of CSNF would be $5.55 billion. Under EPRI Case 1, EPRI estimates that the cost of 5,080 larger capacity TAD canisters would be $3.81 billion, a reduction of $1.74 billion. Under EPRI Case 2, EPRI estimates that the cost of 4,929 larger capacity TAD canisters would be $3.70 billion, a reduction of $1.85 billion.

The YMSEIS calculation package assumed that the unit cost for TAD waste packages would be $600,000 (DOE 2008b, Offsite Manufacturing File, CalcPkg_Manufacturing1_AttchA.xls). Using the scenarios described in Table G-3, under the YMSEIS assumptions, EPRI estimates that the cost of TAD waste packages for disposal of CSNF would be $4.44 billion. Unde EPRI Case 1, EPRI estimates that the cost of 5,080 larger capacity TAD waste packages would be $3.04 billion, a reduction of $1.4 billion. Under EPRI Case 2, EPRI estimates that the cost of 4,929 larger capacity TAD canisters would be $2.96 billion, a reduction of $1.48 billion. As shown in Table G-3, the overall cost savings associated with the use of higher capacity TAD designs would be $3.14 billion under the assumptions in EPRI Case 1 and $3.33 billion under the assumptions in EPRI Case 2.

Table G - 3
Estimated Costs Associated with Disposal of CNSF in TAD Canisters

Scenario Description	Number of TADs	TAD Canister Cost (Billions $)	Waste Package Cost (Billions $)	Tota Cost (Billions $)
DOE YMSEIS (70,000 MTU)				
TADs Loaded at Reactors	6,499	$4.87	$3.90	
TADs Loaded at Repository	901	$0.68	$0.54	
Total Cost		$5.55	$4.44	$9.99
EPRI Case 1				
TADs Loaded at Reactors	4,591	$3.44	$2.75	
TADs Loaded at Repository	489	$0.37	$0.29	
Total Cost		$3.81	$3.04	$6.85
EPRI Case 2				
TADs Loaded at Reactors	4,928	$3.70	$2.96	
TADs Loaded at Reposiotry	1	$0.00	$0.00	
Total Cost		$3.70	$2.96	$6.66

G.4 References

DOE 2008a. Final Supplemental Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada, DOE/EIS-0250F-S1, LSN #: DEN001593669.

DOE 2008b. Yucca Mountain Supplemental Environmental Impact Statement: Calculation Packages in Support of Final Supplemental Environmental Impact Statement for a Geologic

Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada, DOE/EIS-0250F-S1.

DOE 2008c. U.S. DOE, OCRWM, Analysis of the Total System Life Cycle Cost (TSLCC) of the Civilian Radioactive Waste Management Program, DOE/RW-0591, July 2008.

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