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.

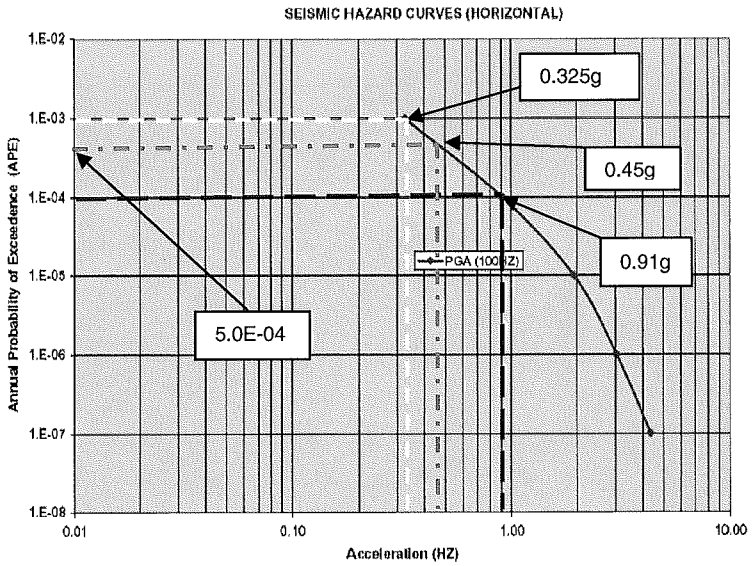

Figure 7 - 1
DOE Horizontal Seismic Hazard Curve [adapted from DOE 2008b, Figure 1.1-89]

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.

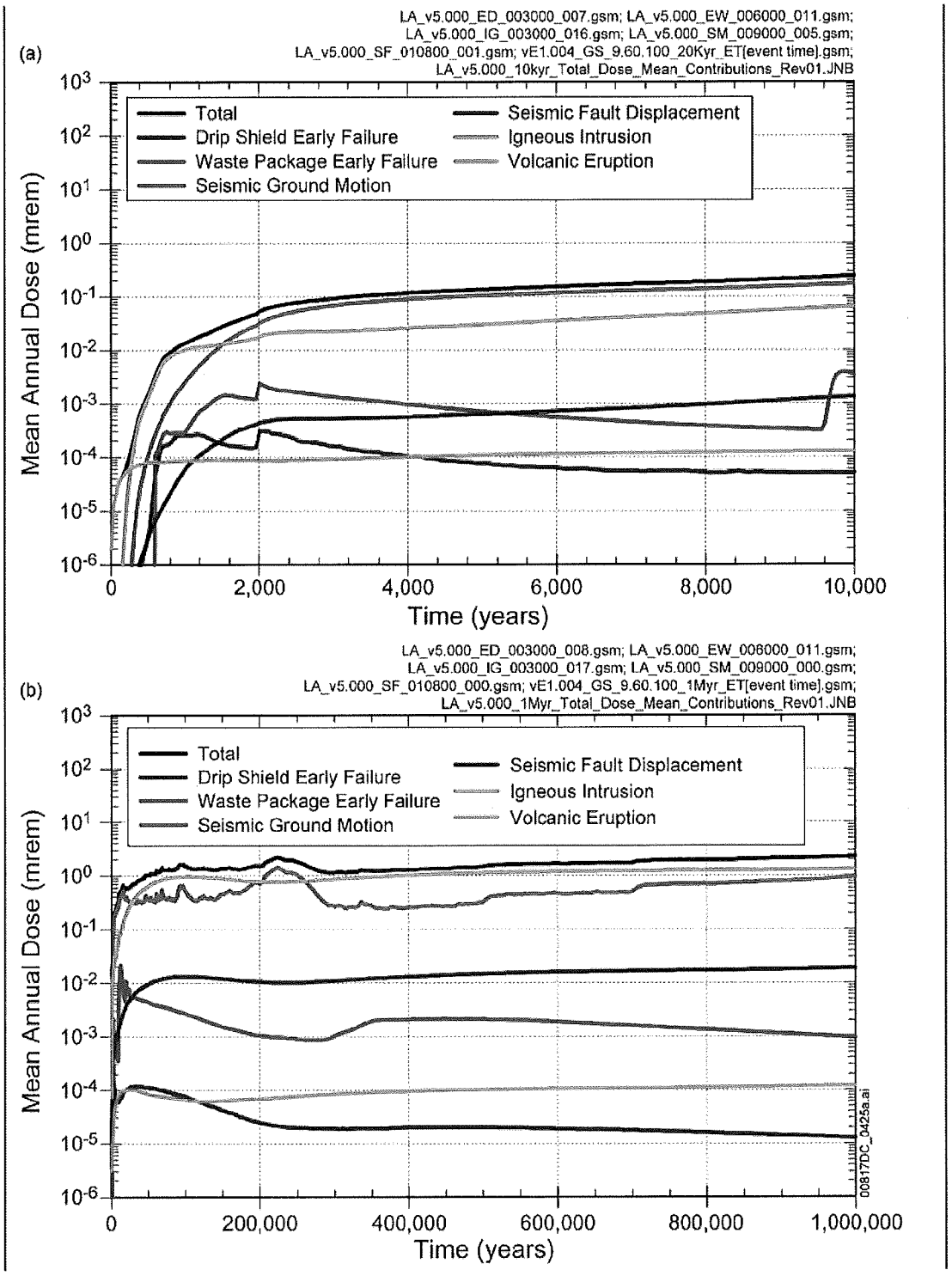

Figure 8 - 1
DOE estimates of RMEI mean annual dose for the 0 to 10,000- (upper figure) and 0 to 1,000,000-year (lower curve) time frames (taken from Figure ES-58 in DOE, 2008b).

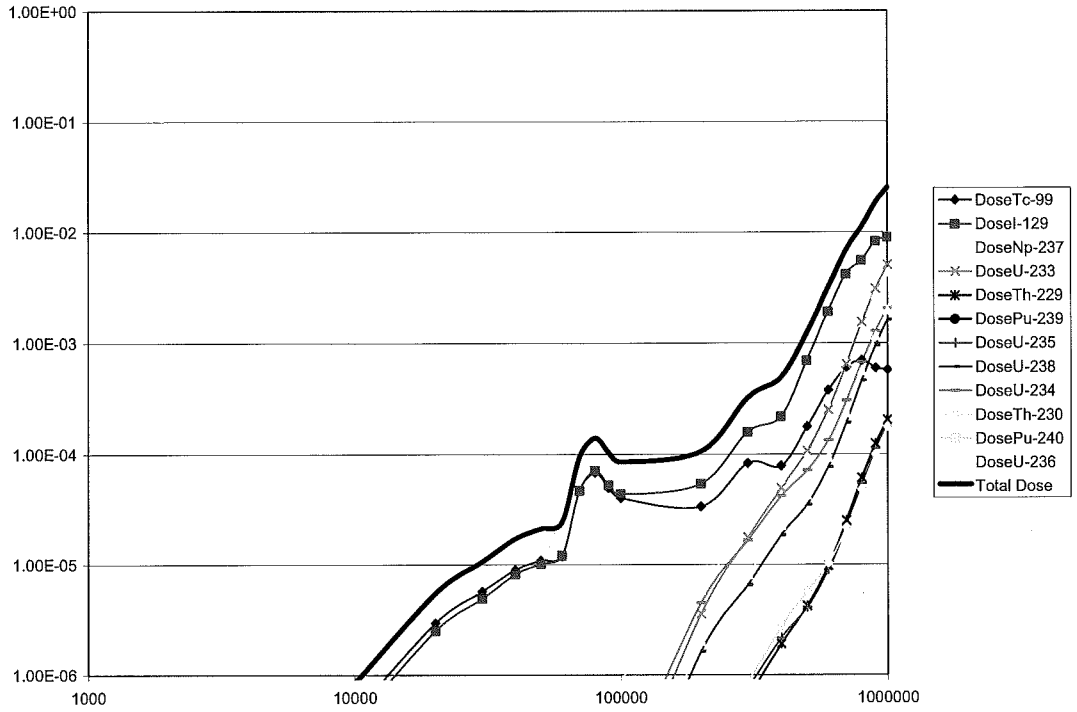

Figure 8 - 2
EPRI CSNF waste package dose results for the Base Case + Seismic Scenarios. Axis units: x: years after repository closure; y: RMEI annual dose rate (mrem).

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)	346c
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)	66,208d
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

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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	▪ 400
▪ Truck Casks	▪ 432
Cask Transfer from ISFSI to Rail Cask	
▪ DPC	▪ 663
▪ TAD	▪ 663
Cask Unloading Operations	
▪ Storage Only Systems	▪ 260
▪ DPC and Dual Purpose Casks	▪ 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