Table 4-3. Summary of Risk Insights Rankings: Significance to Waste Isolation	
ENG1 - Degradation of Engineered Barriers	
Persistence of a Passive Film	High Significance
Waste Package Failure	Medium Significance
Drip Shield	Medium Significance
Stress Corrosion	Medium Significance
Juvenile Failures of the Waste	Low Significance
ENG2 - Mechanical Disruption of Engineered Barriers	
Effects of Accumulated Rockfall on Engineered Barriers	Medium Significance
Dynamic Effects of Rockfall on Engineered Barriers	Low Significance
Effects of Seismic Loading on Engineered Barriers	Medium Significance
Effects of Faulting on Engineered Barriers	Low Significance
ENG3 - Quantity and Chemistry of Water Contacting Engineered Barriers and Waste Forms	
Chemistry of Seepage Water	High Significance
ENG4 - Radionuclide Release Rates and Solubility Limits	
Waste Form Degradation Rate	Medium Significance
Cladding Degradation	Medium Significance
Solubility limits	Medium Significance
Mode of Release from Waste Package	Low Significance
Effect of Colloids on Waste Package Releases	Medium Significance
Invert Flow and Transport	Low Significance
Criticality	Low Significance
UZ1 - Climate and Infiltration	
Present-day Net Infiltration Rate	Medium Significance
Long-term Climatic Change	Medium Significance
UZ2 - Flow Paths in the Unsaturated Zone	
Seepage	High Significance
Hydrologic Properties of the Unsaturated	Medium Significance
Transient Percolation	Low Significance
UZ3 - Radionuclide Transport in the Unsaturated Zone	
Retardation in the Calico Hills non-welded vitric unit	Medium Significance
Matrix Diffusion in the Unsaturated	Medium Significance
Effect of Colloids on Transport in the Unsaturated Zone	Medium Significance
SZ1 - Flow Paths in the Saturated Zone	
Saturated Alluvium Transport Distance	Medium Significance
SZ2 - Radionuclide Transport in the Saturated Zone	
Retardation in the Saturated Alluvium	High Significance
Matrix Diffusion in the Saturated Zone	Medium Significance
Effect of colloids on Transport in the Saturated Zone	Medium Significance
DIRECT1 - Volcanic Disruption of Waste Packages	
Probability of Igneous Activity	High Significance
Number of Waste Packages Affected by Eruption	High Significance
Number of Waste Packages Damaged by Intrusion	Medium Significance
DIRECT2 - Airborne Transport of Radionuclides	
Volume of Ash Produced by an Eruption	Medium Significance
Remobilization of Ash Deposits	Medium Significance
Inhalation of Resuspended Volcanic Ash	High Significance
Wind Vectors During an Eruption	Medium Significance

Table 4-3. Summary of Risk Insights Rankings: Significance to Waste Isolation (continued)	
DOSE1 - Concentration of Radionuclides in Ground Water Well-pumping Model	Low Significance
DOSE2 - Redistribution of Radionuclides in Soil Redistribution of Radionuclides in Soil	Low Significance
DOSE3 - Biosphere Characteristics Characterization of the Biosphere	Low Significance

Figure 4-4 shows the calculated dose assuming that: (i) in the basecase, passive conditions prevail for all waste packages; and (ii) passive conditions are not maintained for 25 percent of the waste packages. In this latter case, the expected dose approaches 0.01 mSv/yr [1 mrem/yr] after 10,000 years. As a result of the low corrosion rates under passive conditions (basecase), the first waste package failure from uniform corrosion is estimated to occur after the 10,000-year performance period. Doses that occur before 10,000 years in the basecase estimate are the result of juvenile waste package failure. Assuming a uniform corrosion rate in the range from 5.0×10^{-5} to 5.4×10^{-4} mm/yr [2.0×10^{-6} to 2.1×10^{-5} in/yr], initial waste package failures from corrosion occur after 37,000 years and all waste packages fail after 403,000 years. The absence of passivity is assumed to result in uniform corrosion rates ranging from 5.0×10^{-3} to 5.4×10^{-2} mm/yr [2.0×10^{-4} to 2.1×10^{-3} in/yr]. Failure times for waste packages in the absence of a protective, stable passive film are estimated to range from approximately 400 to 4,000 years.

Uncertainties

The corrosion rate of the alloys proposed for the engineered barrier system, such as Alloy 22 and titanium alloys, are controlled by the presence of a thin, protective oxide film that restricts metal dissolution. The stability of oxide films on passive alloys is dependent on the material and exposure conditions. Loss of passivity can lead to corrosion rates that are orders of magnitude greater than those measured under passive conditions. For chromium-containing alloys such as Alloy 22, loss of passivity (depassivation) can occur in aggressive solutions characterized by low pH and high chloride concentrations, especially at high temperatures. See Section 4.3.3 for a discussion of the likelihood of such conditions. Chromium-containing alloys exhibit transpassive dissolution at high anodic potentials, also resulting in high corrosion rates. Other processes, such as anodic segregation of sulfur or preferential dissolution of alloying elements, may also disrupt passivity. Passive films on titanium alloys are very stable in chloride solutions, but are strongly affected by the presence of fluoride, which significantly enhances the uniform corrosion rate (Brossia, et al., 2001). As discussed in Section 4.3.3, there are uncertainties associated with the amount and concentration of fluoride in the water seeping into the drift.

Minor increases in fluoride concentration in pore water may have a significant effect on corrosion rate.

Waste Package Failure Mode: Medium Significance to Waste Isolation

The failure mode of the waste package (uniform corrosion, localized corrosion, or stress corrosion cracking) and its morphology (e.g., pits, cracks, or large corrosion holes or patches) is important for determining the amount of water that can enter the waste package.

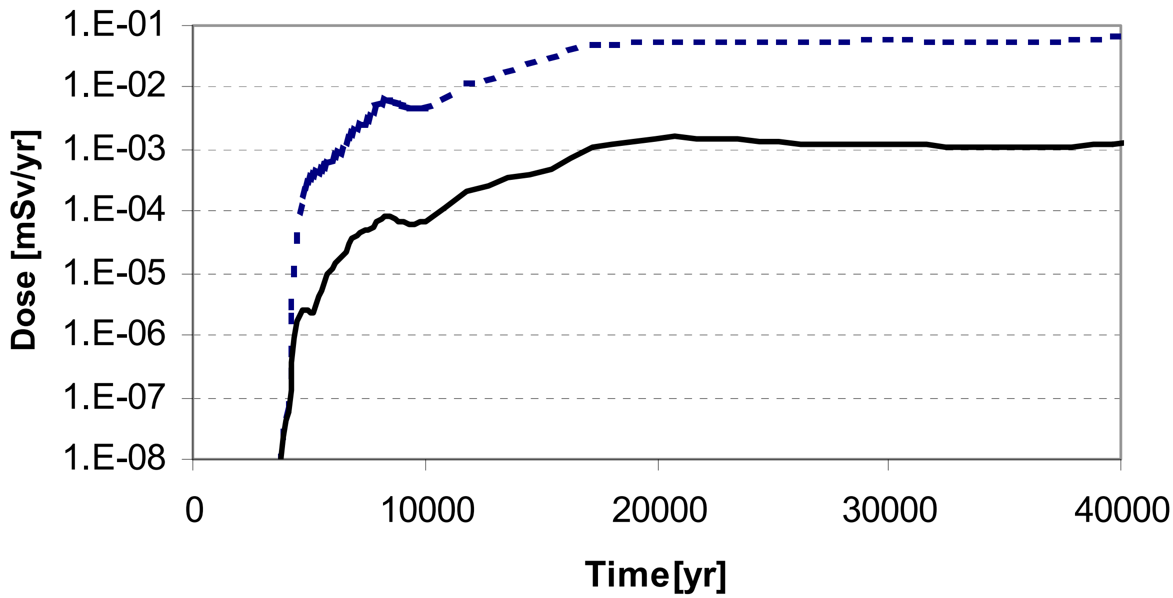

Figure 4-4. Dose Calculations as a Function of Waste Package Corrosion Rate. In the Basecase (TPA Version 4.1j Code) (Continuous Line), all Waste Packages Are Assumed to Undergo Passive Dissolution. Assuming Enhanced General Corrosion Rate for 25 Percent of the Waste Packages (Broken Line) Results in an Increase in the Calculated Dose.

Discussion

Different corrosion mechanisms create different types of failures (openings) in the surfaces of the engineered system. The amount of water that will enter a waste package and the amount of waste that will be transported out of a waste package will be influenced by the size of the openings. Intact surfaces will divert water away from the waste. The failure modes and morphology only have significant influence on the risk estimate when the openings are of limited surface area and frequency or when the waterflow rates are very small. The degree of pessimism introduced into the analyses can reduce the effectiveness of geometrical considerations to limit the release of radionuclides. For diffusional releases, the size of the openings at which there is no longer a significant limitation on mass transport will be influenced by assumptions for boundary conditions and flow paths before radionuclides arrive at the opening and after they exit the opening. Current understanding indicates that stress corrosion cracks and pits that would form from localized corrosion will likely be small in cross-sectional area such that capillary forces may strongly limit the advective transport of water and radionuclides through such openings.

Alloy 22 was selected as the waste package outer container material to mitigate degradation and failure that can result from stress corrosion cracking, hydrogen embrittlement, localized corrosion, and accelerated uniform corrosion as a consequence of loss of passivity (CRWMS M&O, 2000a). Under conditions where the passive film can be maintained, failure of the waste package by corrosion-only processes is unlikely within the 10,000-year compliance period. Although highly resistant to various modes of corrosion, Alloy 22 is susceptible to localized corrosion, stress corrosion cracking, and hydrogen embrittlement.

Crevice corrosion of Alloy 22 can occur if aggressive solutions are in contact with the waste package and the corrosion potential exceeds the critical potential (i.e., the repassivation potential) for localized corrosion (Brossia, et al., 2001; Dunn, et al., 2003). In addition to the nature of the metal or alloy, the corrosion potential is mainly dependent on temperature and solution pH, whereas the repassivation potential for localized corrosion is dependent on temperature, the concentrations of aggressive and inhibiting species, and the microstructure of the material. Figure 4-5 shows the repassivation potential for Alloy 22 as a function of material condition and chloride concentration and the corrosion potential as a function of pH (Dunn, et al., 2003). In concentrated chloride solutions with a low pH, localized corrosion can be initiated when the corrosion potential exceeds the repassivation potential for localized corrosion. Localized corrosion is not expected in solutions with low chloride concentrations at neutral or alkaline pH, because the corrosion potential is below the repassivation potential. The localized corrosion susceptibility is also dependent on the relative concentrations of inhibiting and aggressive species. No localized corrosion will occur if a sufficient concentration of inhibiting species is present in the water contacting the waste package.

Figure 4-6 shows the expected dose for the basecase and conditions where 10 percent of the waste packages are exposed to aggressive environmental conditions that promote localized corrosion. In the latter case, the peak expected dose is on the order of 0.001 mSv/yr [0.1 mrem/yr] during the first 10,000 years. The calculations shown in Figure 4-6 assume a limited amount of water can enter the waste package after corrosion penetrates the Alloy 22 outer container, which accounts for failure of the waste package from all corrosion processes.

Failure of the engineered barrier system components by uniform corrosion may result in large openings in the waste package. In contrast, pitting and crevice corrosion will likely result in small localized penetrations of the waste package. Stress corrosion cracks may also have a small aperture that will limit radionuclide release. The effects of localized penetrations of the waste package may be influenced by the combined effects of mechanical loads that are likely to occur as a consequence of rockfall, drift degradation, and seismic events. Mechanical loading of waste packages that are degraded as a consequence of uniform or localized corrosion, stress corrosion cracking, or hydrogen embrittlement may increase the size of the failure openings detrimentally affecting overall system performance.

Uncertainties

The amount of water that will enter a waste package and the amount of waste that will be transported out of a waste package will be influenced by the size of the openings. Release of radionuclides, in particular via diffusive mechanisms, will correlate with the failed surface area. Degradation modes that may lead to failure of the engineered barrier system and allow release of radionuclides include corrosion processes such as uniform and localized corrosion, stress corrosion cracking, and hydrogen embrittlement and mechanical interactions as a result of disruptive events. The location, size, and orientation of failure openings will be influenced by

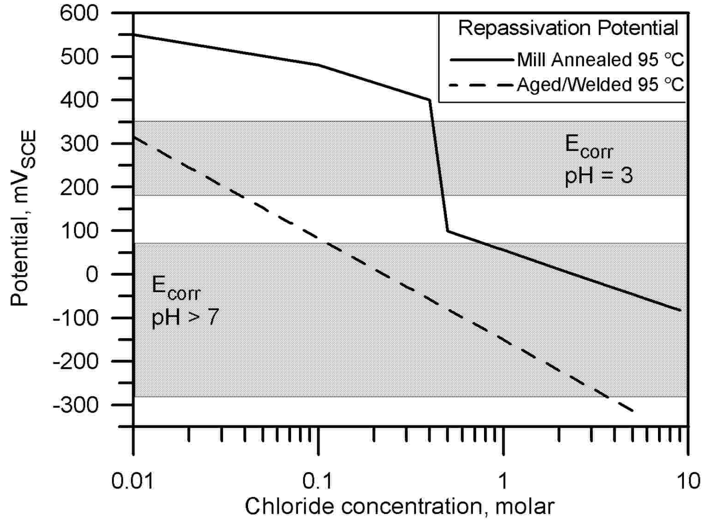

Figure 4-5. Repassivation Potential for Crevice Corrosion for Mill-Annealed and Thermally Aged or Welded Alloy 22 as a Function of Chloride Concentration at 95 °C [203 °F]. Range of Corrosion Potentials as a Function of pH are Shown as Shaded Banks. (Dunn, et al., 2003)

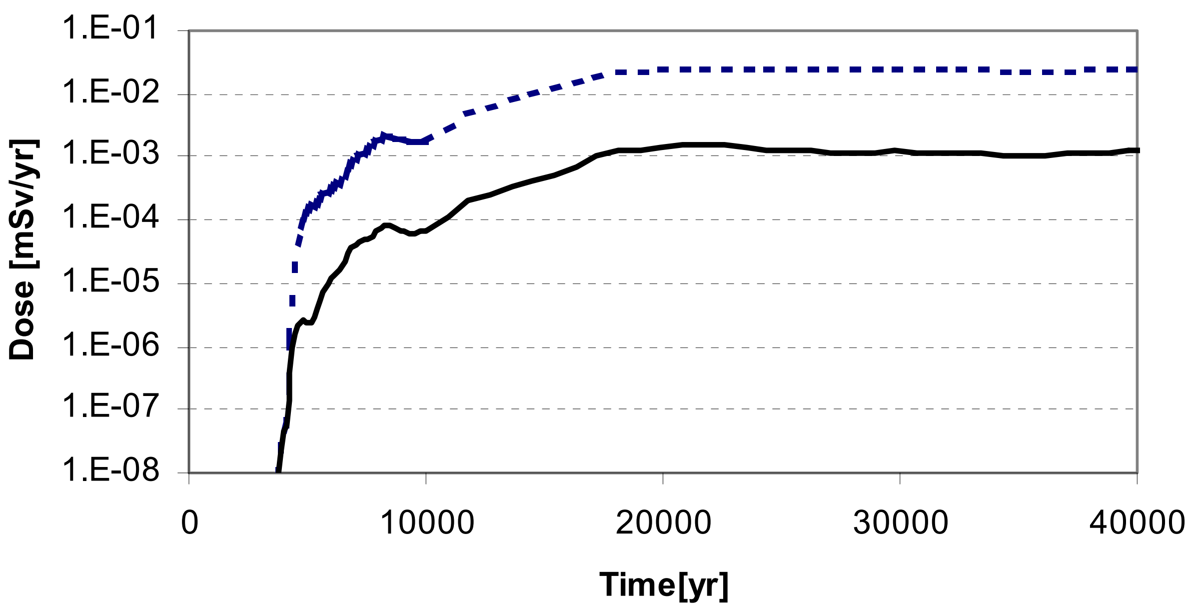

Figure 4-6. Effect of Waste Package Corrosion Mode on Calculated Dose. All Waste Packages are Assumed to Undergo Passive Dissolution in the Basecase (Cotninuous Line). Assuming 10 Percent of the Waste Packages Are Exposed to Aggressive Sol.utions that Promote Localized Corrosion Increases the Calculated Dose (Broken Line).

corrosion processes and mechanical interactions. However, there are uncertainties related to the specific characteristics of these openings, depending on the area of contact with water, the presence of deposits on the waste package surface, and the action and nature of the applied loads.

Drip-Shield Integrity: Medium Significance to Waste Isolation

The integrity of the drip shield will influence the quantity and chemistry of the water that can develop on the waste package and the potential effects on corrosion modes and rates.

Discussion

The quantity and chemistry of the water contacting the waste package and the drip shield is addressed in Section 4.3.3 of this report. Analyses performed by DOE for the total system performance assessment for site recommendation (Figure 4-7) show the drip shield has little effect on repository performance (CRWMS M&O, 2000b). However, the role of the drip shield to control the formation of aggressive environments on the waste package surface was not included in the DOE model. Figure 4-8 shows the effect of drip-shield integrity on the estimated dose, assuming 10 percent of the waste packages are exposed to environments that promote localized corrosion. Higher doses observed with accelerated drip-shield failure are attributed to water seepage, which is not diverted by the drip shield, and hence contacts the breached waste packages.

Depending on the timing and extent of rockfall, the drip-shield design may be important for limiting damage to the waste package (see Section 4.3.2.1).

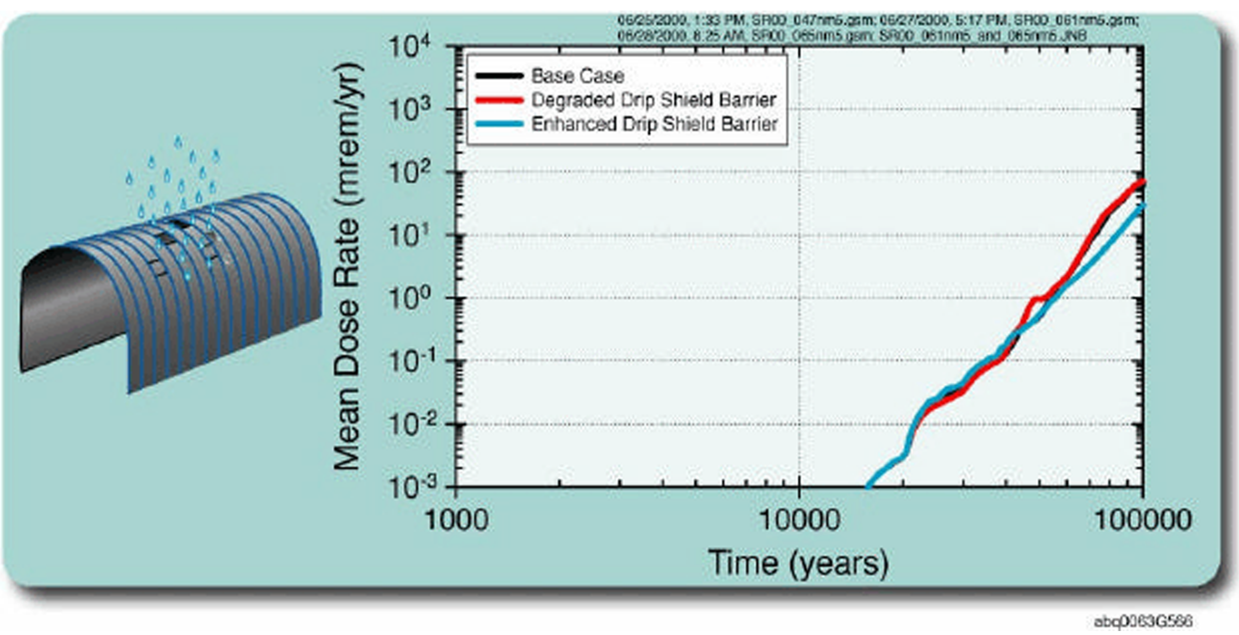

Figure 4-7. Drip-Shield Sensitivity Analysis Using the Total System Performance Assessment for Site Recommendation (CRWMS M&O, 2000b, Figure 5.3-3) (1 mrem/yr = 0.01 mSv/yr)

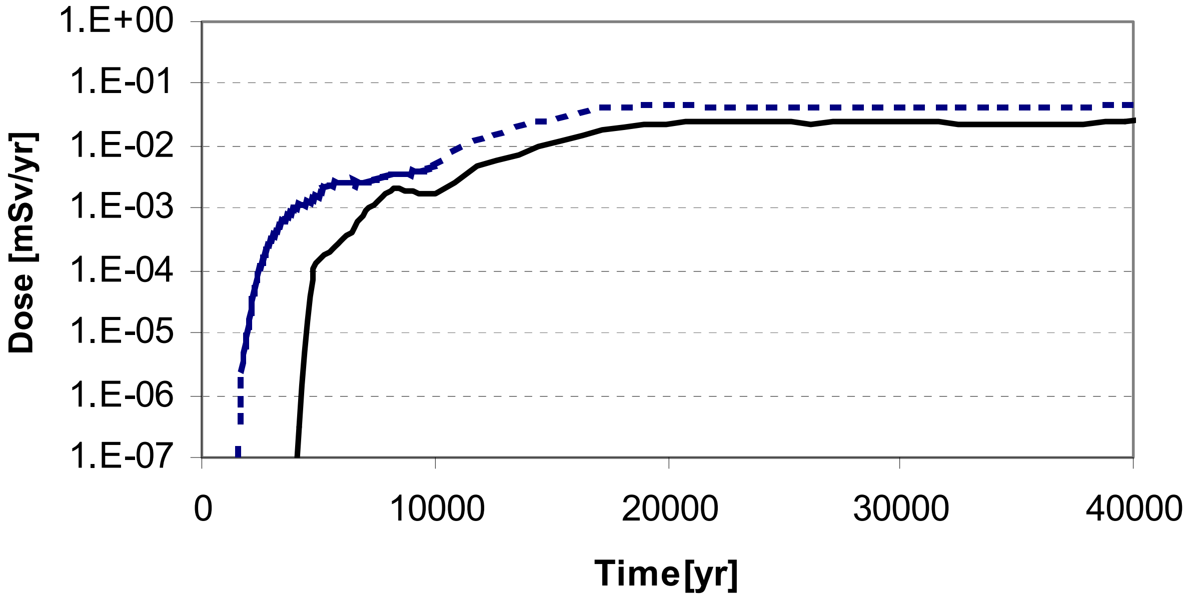

Figure 4-8. Effect of Drip-Shield Failure on Calculated Dose. In the Reference Case (10 Percent of Waste Packages are Assumed to Display Localized Corrosion), Drip-Shield Failures Are Assumed to Have a Lognormal Distribution with 0.1 Percent of Drip Shields Failing at 2,700 Years and 99.9-Percent Failing at 20,400 Years (Continuous Line). Enhanced Failures of Drip Shields Assume a Uniform Distribution with 10 Percent of the Drip Shields Failing Between 1,000 and 5,000 Years (Broken Line).

Uncertainties

The drip shield is intended to limit ground water contact with the waste package. While the drip shield is intact, water that contacts the waste package may be limited to condensed water with low concentrations of aggressive species that are unlikely to promote localized corrosion or stress corrosion cracking. The drip shield will be constructed from titanium alloys that are resistant to localized corrosion in chloride solutions over a wide temperature range. However, the uniform corrosion rate of titanium alloys is dependent on the fluoride concentration. Faster corrosion rates and shorter failure times may occur on drip-shield sections exposed to solutions with fluoride concentrations greater than 10^{-4} molar (M) (Brossia, et al., 2001; Lin, et al., 2003). However, titanium corrosion may be limited by the supply of fluoride from dripping water, and not strictly to the concentration threshold (Lin, et al., 2003). Failure of the drip shield by corrosion degradation mechanisms in combination with mechanical disruption may allow the formation of aggressive environments in contact with the waste package surface and lead to accelerated failure of the waste package. The formation of aggressive environments depends on many factors, with different degrees of uncertainties, that are related to the deposition of deliquescent salts, the rate of evaporation, and the composition of the seepage water. These aspects are discussed in Section 4.3.3.

Stress Corrosion Cracking: Medium Significance to Waste Isolation

Stress corrosion cracking of the drip shield or waste package affects a limited area and, thus, this corrosion process is not expected to allow substantial amounts of water to enter the waste package. However, applied loads arising from accidental internal overpressure, rockfall, or seismic events may increase the failure area and facilitate the ingress of water through the extended opening of stress corrosion cracks.

Discussion

The stress corrosion cracking susceptibility of Alloy 22 is dependent on material condition, corrosion potential, and stress intensity (Andresen, et al., 2003). Crack propagation rates for mill-annealed, cold-worked, and thermally aged Alloy 22 in basic saturated water are shown in Table 4-4. Under constant loading conditions, the crack propagation rates of the mill-annealed alloy decrease with time. However, sustained crack propagation under constant loading conditions has been observed for Alloy 22 in the cold-worked and thermally aged conditions. Recent results reported by General Electric and Lawrence Livermore National Laboratory and results obtained in independent tests conducted at the Center for Nuclear Waste Regulatory Analyses (CNWRA) show that Alloy 22 is susceptible to stress corrosion cracking in simulated concentrated water at temperatures below boiling, provided sufficient stress intensity is present.

The effect of stress corrosion cracking on the estimated dose is likely to be low because of the limited area of the cracks. In Figure 4-9, mean values (from 100 realizations) of radionuclide release rates (I-129, Tc-99, and Np-237) from the waste package are compared to estimates of diffusive release. Case 1 (continuous lines) is the reference case which assumes 90 percent of the waste packages are breached by general corrosion and 10 percent by localized corrosion. Case 2 (lines with circles) considers that radionuclides are released from the waste package

Table 4-4. Crack Propagation Rates for Alloy 22 in Basic Saturated Water

Material Condition	Stress Intensity, $MPa\cdot m^{1/2}$	Crack propagation Rate, mm/s
Mill-annealed	30	0
	45	4×10^{-10} — 1.3×10^{-9}
Mill-annealed plus 20% cold work	30	2×10^{-10} — 5×10^{-10}
Thermally aged 175 hours at 700 °C (1292°F)	16.5	—
	24.2	8×10^{-10} — 1.3×10^{-9}

(From Andresen, et al., 2003)

NOTES: English equivalents for $MPa\cdot m^{1/2}$ and mm/s are as follows: $MPa\cdot m^{1/2} \times 0.9091$ = ksi $in^{1/2}$ and mm/s × 0.039 = in/s.

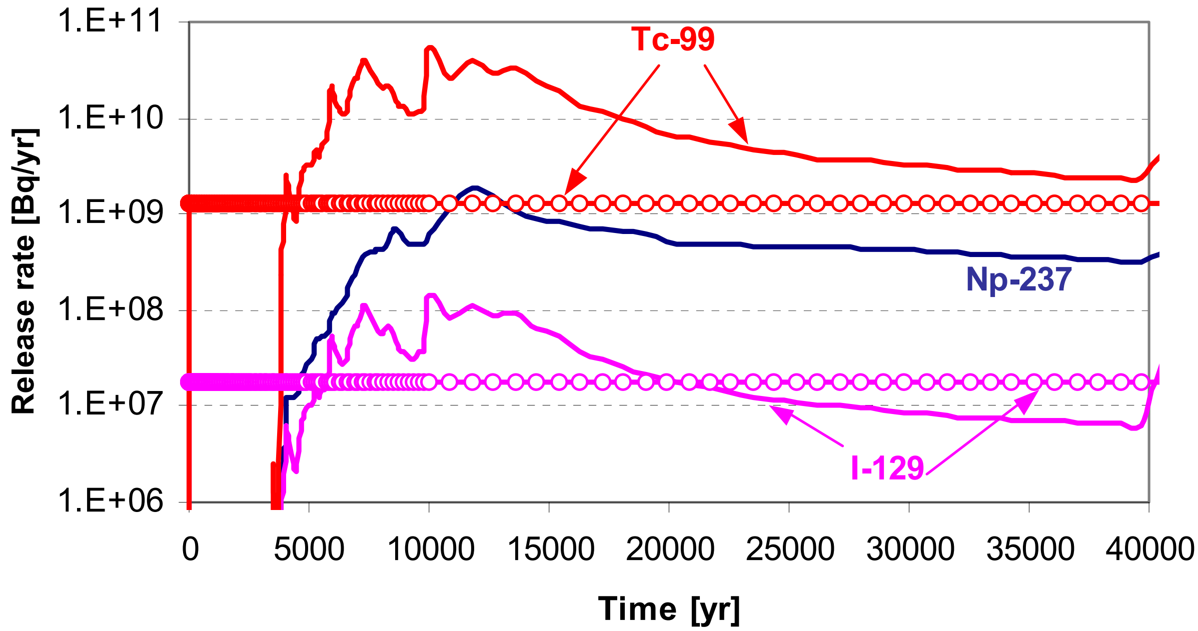

Figure 4-9. Radionuclide Release Rates from the Waste Package for Two Cases. Case 1 (Continuous Lines) Is a Reference Case Accounting for 90 Percent of the Waste Packages Breached by General Corrosion and 10 Percent by Localized Corrosion. Case 2 (Lines with Circles) Shows Diffusive Release Rates (I-129 and Tc-99) Assuming Saturation at the End of the Water Film in Contact with the Spent Nuclear Fuel, and Zero Concentration at the Other End. (Tc-99; I-129; Np-237).

only through stress corrosion cracks by a diffusive mechanism. In deriving these release rates (lines with circles), the following assumptions were made: (i) all the waste packages are breached by stress corrosion cracking at the time of emplacement; (ii) a thin film of water connects the spent nuclear fuel with the exterior of the waste package; (iii) the radionuclide concentration in the film, at the point of contact with the spent nuclear fuel, is determined by the saturation of radionuclide-bearing solids; (iv) the concentration at the end of the diffusive path is zero; (v) the problem is one-dimensional with a path length equal to 0.3 m [1.0 ft] and film cross section of 10^{-8} m^2 [10^{-7} ft^2]; and (vi) no credit is taken for cladding protection. As seen in Figure 4-9, diffusive releases, as modeled, are dominant during the first 4,000 years; however, they are at least one order of magnitude below the maximum release rates of Case 1 (I-129 and Tc-99), occurring at around 10,000 years. The estimated diffusive releases of Np-237 are less than 10^5 becquerels/yr [10^{-5} curies/yr] and are not displayed in Figure 4-9. After approximately 4,000 years, the radionuclide release rates associated with Case 1 exceed the diffusive release rates.

Advective release through stress corrosion cracks could occur if cracks were opened by mechanical loading (e.g., mechanical interactions of the waste package with other components of the engineer barrier system during seismic events or as a result of static loading). Figure 4-10 shows the estimated effects of advective release of Np-237, I-129, and

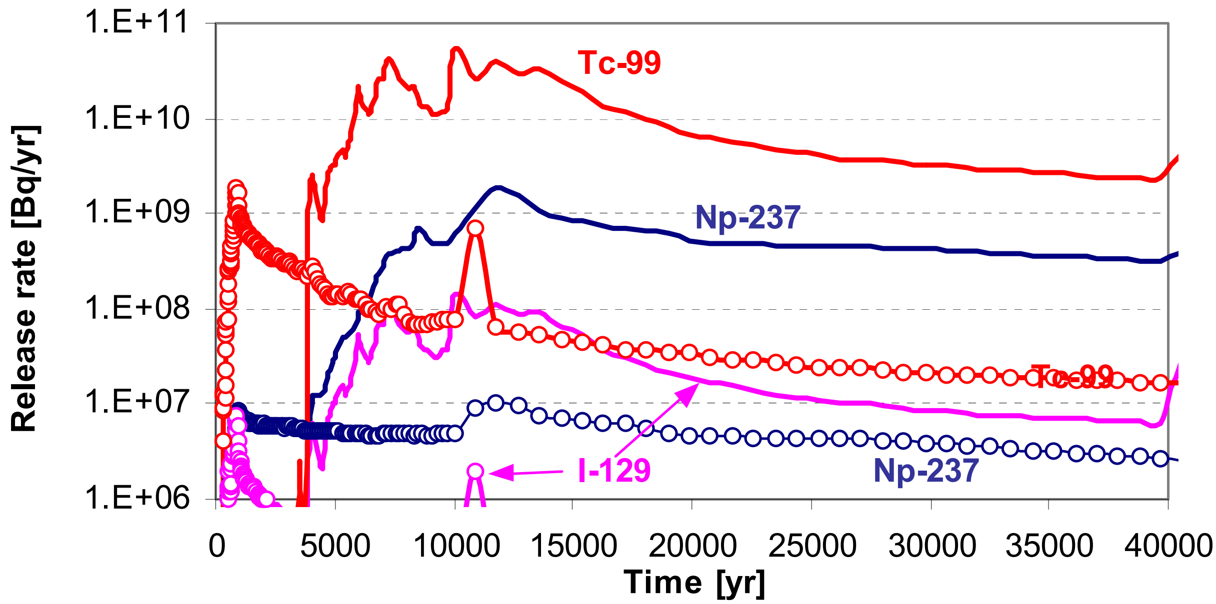

Figure 4-10. Radionuclide Release Rates from the Waste Package for Cases 1 and 3. Case 1 (Continuous Lines) Is a Reference Case Accounting for 90 Percent of the Waste Packages Breached by General Corrosion and 10 Percent by Localized Corrosion. Case 3 (Lines with Circles) Shows Advective Release Rates Estimated by Multiplying the Waste Package Release Rates in the Flow-Through Scenario by a Factor of 1/1,000 (Factor to Account for Protection Against Seepage Offered by Unbreached Waste Package Surface). Both Curves Correspond to Mean Values from 100 Realizations (TC-99, I-129, and Np-237).

Tc-99 through cracks. To facilitate comparison, Figure 4-10 includes radionuclide release rates of Case 1 from Figure 4-9. Stress corrosion cracks were assumed to develop during a period of temperatures above the boiling point of water and relative humidity above a deliquescence point of salt formation. Radionuclide release rates from the waste package were estimated for the flow-through scenario, and decreased by a factor of 1/1000 to account for protection against seepage offered by the unbreached area of the waste package. Protection by the drip shield or cladding was disregarded. Under these assumptions, it was estimated that advective release through stress corrosion cracks was largest during early years of repository operation, up to 4,000 years (Case 3, lines with circles in Figure 4-10). However, maximum release rates of Np-237, Tc-99, and I-129 are one order of magnitude lower than those derived for Case 1 (continuous lines).

<u>Uncertainties</u>

Stress corrosion cracking requires the combination of a susceptible material or microstructure, an aggressive environment, and an applied or residual tensile stress. Although nickel-base alloys are known to be resistant to environmentally assisted cracking in hot chloride solutions, stress corrosion cracking of Alloy 22 has been reported in simulated ground water solutions that may contact the waste packages (Andresen, et al., 2003; King, et al., 2002; Estill, et al., 2002).

Stress corrosion cracks that penetrate the waste package may be tight and may restrict the transport of water into the waste package. Cracks that remain tight may allow only diffusive release of radionuclides. Stress corrosion cracking coupled with mechanical loading from disruptive events may propagate existing cracks or enlarge existing failures and allow advective release of radionuclides.

There is significant uncertainty associated with crack geometry and the effect of applied loads arising from accidental internal overpressure, rockfall, or seismic events that may propagate existing stress corrosion cracks, or lead to mechanical failure of the degraded waste package.

Release rates associated with stress corrosion cracking may be influenced by: (i) limited cross section of stress corrosion cracks, (ii) frequency of nucleation sites for stress corrosion cracks, (iii) frequency of chemical solutions necessary for stress corrosion cracking, (iv) levels of material stress needed for stress corrosion cracking, (v) stress corrosion cracking propagation rate, and (vi) crack location on the waste package geometry.

Juvenile Failures of the Waste Package: Low Significance to Waste Isolation

Juvenile or early failures of the waste package (e.g., closure welding defects, such as flaws, which can promote other degradation processes) are expected to be limited to a small fraction of waste packages and not have a significant effect on waste package performance and hence on radionuclide release.

Discussion

The basecase results from the TPA code, and the results from the DOE total system performance assessment for the supplemental science and performance analyses model, quantitatively support that juvenile or early failures of the waste package are of low significance to waste isolation (Figures 4-11 and 4-12). For the TPA Version 4.1 basecase, the risk from the nominal scenario (on average 44 juvenile failures, or 0.63 percent) is 0.00021 mSv/yr (0.021 mrem/yr) at 10,000 years. The DOE total system performance assessment for the supplemental science and performance analyses model had on average less than one package failure per stochastic realization, and the resultant doses were very small at 10,000 years [e.g., less than 10^{-6} mSv/yr (10^{-4} mrem/yr)]. Quality assurance procedures for fabrication, characterization, handling, and emplacement of waste packages should reduce the likelihood of significant defects and, therefore, juvenile failures. Quantitative analyses to date demonstrate the repository system is likely to tolerate limited waste package failures. Processes such as loss of passivity or occurrence of localized corrosion are not considered to be part of juvenile failures and are modeled separately.

Uncertainties

Initial defects coupled with waste package degradation processes and mechanical loading as a result of disruptive events may lead to early failures of waste packages. The number of waste packages that are susceptible to early failure processes will be dependent on the frequency, type, size, and orientation of the initial defects.

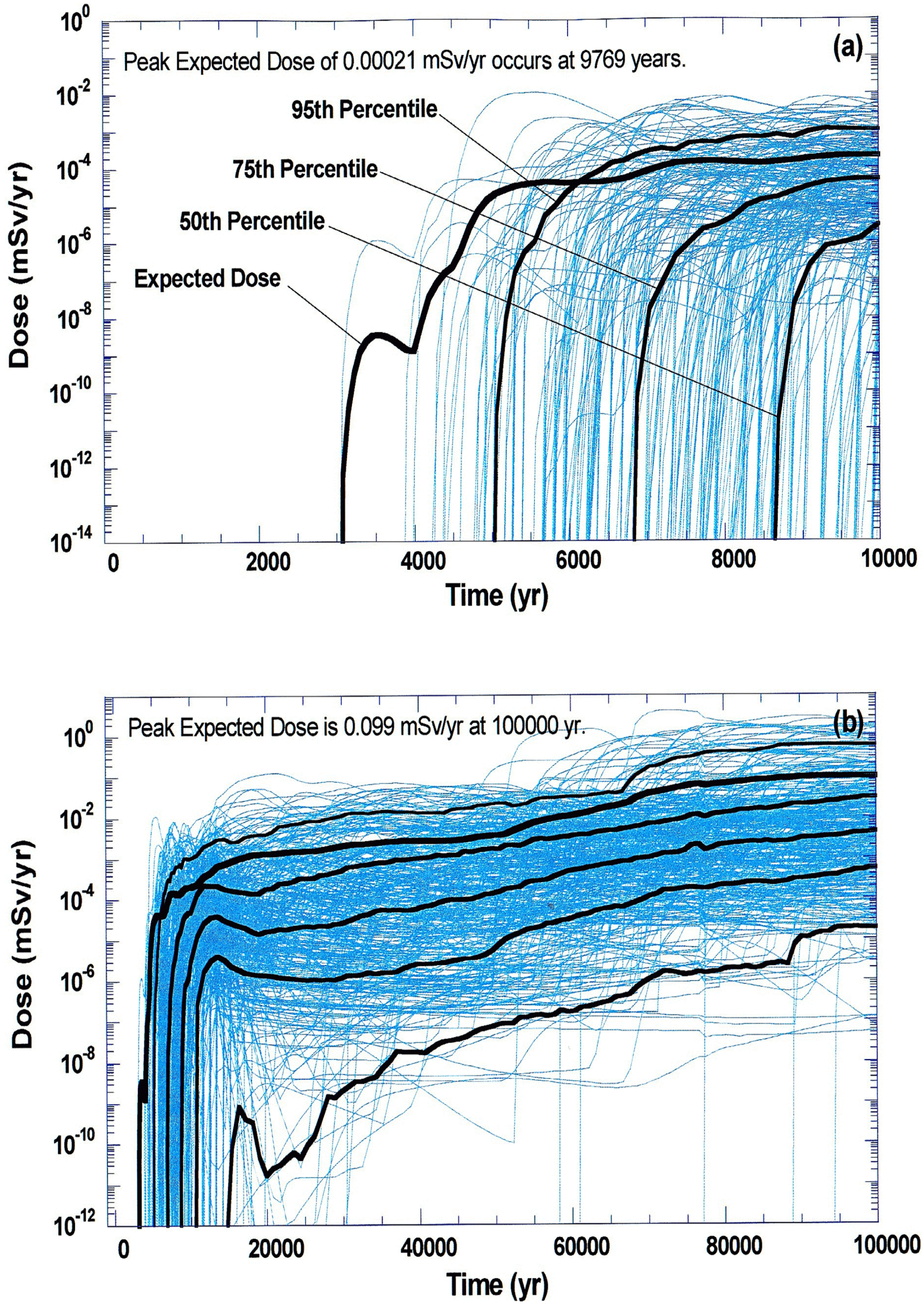

Figure 4-11. Ground Water Dose in (a) 10,000 Years, and (b) 100,000 Years, Including the Average Dose for 150 Realizations (Mohanty, et al., 2002, Figure 3-20)

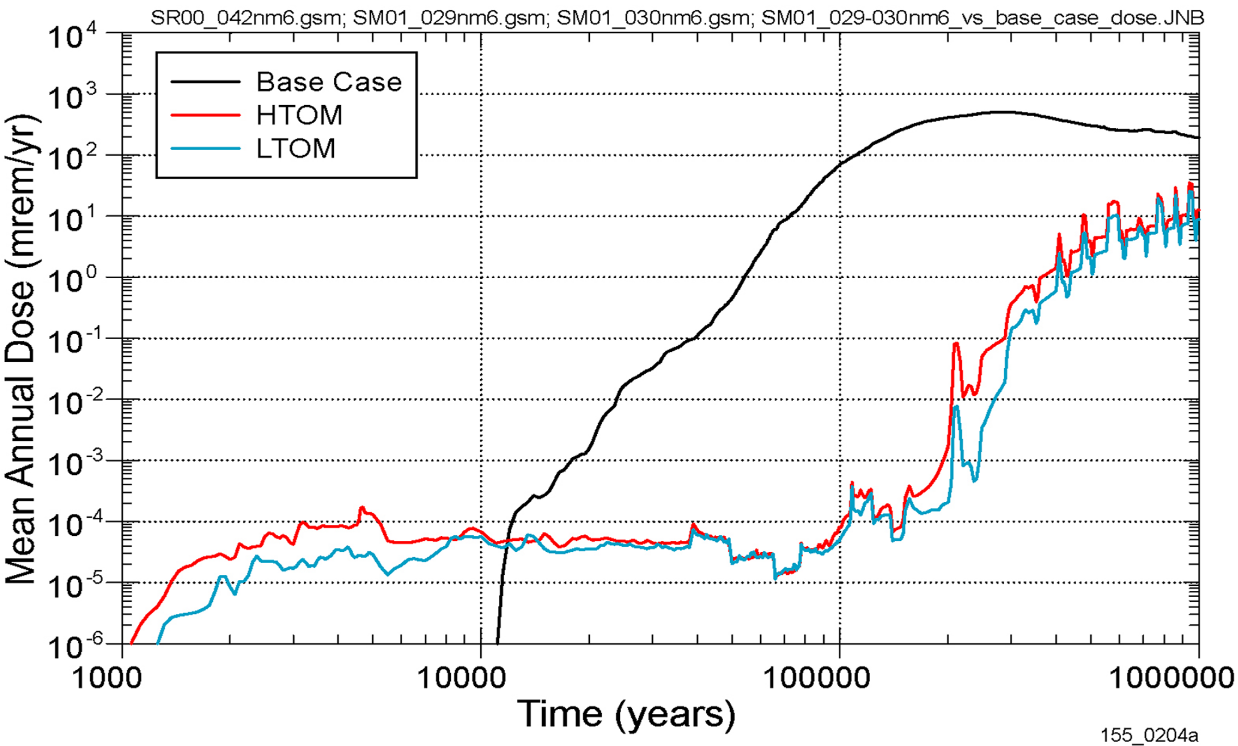

Figure 4-12. Summary of Peak Dose Performance Results (HTOM: High-Temperature Operating Mode; LTOM: Low-Temperature Operating Mode; Basecase: TSPA-SR) (Bechtel SAIC Company, LLC, 2001, Figure 4.1-1) (1 mrem/yr = 0.01 mSv/yr)

4.3.2 Mechanical Disruption of Engineered Barriers (ENG2)

Risk Insights:	
Effects of Accumulated Rockfall on Engineered Barriers	Medium Significance
Dynamic Effects of Rockfall on Engineered Barriers	Low Significance
Effects of Seismic Loading on Engineered Barriers	Medium Significance
Effects of Faulting on Engineered Barriers	Low Significance

4.3.2.1 Discussion of the Risk Insights

Effects of Accumulated Rockfall on Engineered Barriers: Medium Significance to Waste Isolation

Mechanical loading from rockfall rubble accumulated from drift degradation over time may lead to failure of the drip shields and waste packages. The failure of the drip shields and waste packages will depend on the rate of accumulation of rockfall rubble in the drift (building static load on the drip shield) and the threshold load-bearing capacity of the drip shields and the waste packages. The accumulation of rock rubble in the drift outside the drip shield will also increase the waste package and drip-shield temperatures.

Discussion

Current understanding of the degradation of mined openings indicates that all of the drip shields could experience static loads from rockfall rubble that can accumulate after repository closure. These loads are expected to damage the drip shields. A total system performance assessment calculation (Figure 4-13) indicates that if all drip shields fail from rockfall without any associated waste package failure, the dose consequence will be low. A calculation in which all drip shields are assumed to fail simultaneously at the beginning of the postclosure period increased the peak expected dose by less than 75 percent relative to the nominal scenario, which is still at least two orders of magnitude below the regulatory limit. The higher temperature associated with rockfall and the potential increase in the amount of water entering the drift are not expected to significantly increase this result.

Sustained rockfall rubble loads may cause some of the drip shields and waste packages to fail. At the present time, however, the TPA code does not have a model to compute mechanical failure of the waste package if the drip shield is no longer capable of isolating it from rockfall loads. To address this interim limitation, another total system performance assessment calculation was performed to bound the potential effects of both the drip shields and waste packages failing at the time of repository closure. The results of this analysis (Figure 4-13) indicate that a simultaneous failure of all drip shields and waste packages increases dose significantly above the nominal scenario dose, which has a limited number of waste packages (i.e., 40 on average) failing before 10,000 years. However, the dose is below the regulatory dose limit, and the timing and extent of drift collapse is highly uncertain. This uncertainty has

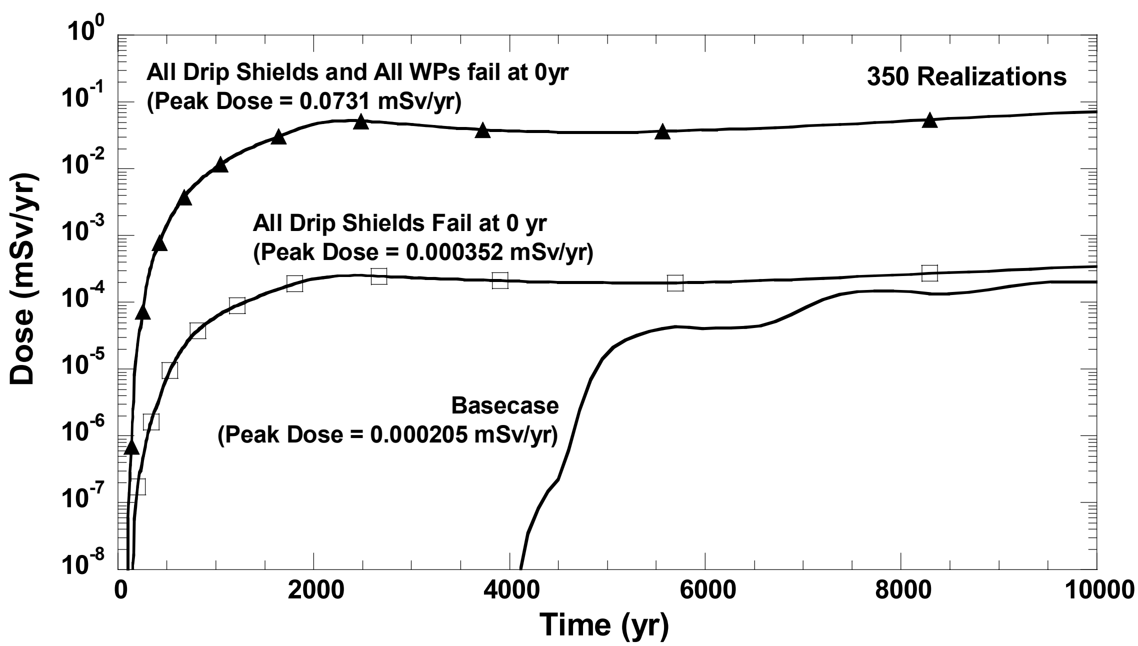

Figure 4-13. Conditional Peak Expected Doses Corresponding to (i) the Basecase, (ii) a Hypothetical Case in which all Drip Shields Have Failed at Postclosure, and (iii) a Hypothetical Case in which all Drip Shields and Waste Packages Have Failed at the Time of Postclosure (Doses Have Not Been Weighted by the Probability of Scenario Occurrence

implications that the significance associated with the process could range from minimal to high, but is expected to be less than the estimates of this bounding analysis. Additionally, a drip-shield design that limits damage to waste packages would limit the significance of drift collapse.

The insulating effects of the rockfall rubble will increase drip-shield, waste-package, and waste-form temperatures. High temperatures will adversely affect the load-bearing capacity of the drip shield and the waste package, thus increasing their failure potential during the duration of high temperatures. The increased temperature also may accelerate drip-shield and waste-package corrosion and waste-form dissolution.

DOE calculations with engineered backfill, which can be viewed as an upperbound for natural backfilling with rockfall rubble, indicate that the peak waste package temperature could increase nearly two fold to 315 °C [600 °F] , compared to the no-backfill scenario (DOE, 2000; p. 90). However, backfill from drift degradation is anticipated to contain more void space during times relevant to the repository thermal pulse than engineered backfill, thereby, limiting peak waste package temperature. The temperature of other components of the engineered system could be affected correspondingly. It is unlikely that liquid water will be present within the drift at temperatures above 160 °C [320 °F]. The NRC temperature estimates for the unbackfilled condition and the DOE temperature estimates for engineered backfill bound the time for which temperatures are elevated. The analyses estimate that drift degradation is not expected to substantially increase the length of time that engineered barrier system components remain above the critical temperature threshold for the occurrence of localized corrosion. Therefore, the effect on corrosion may not be substantially different from the nominal no-backfill scenario. NRC performed an analyses to determine the effect of increased temperature as a result of backfill from drift degradation on waste-form dissolution. The use of a higher-dissolution rate model in which the waste form dissolves in less than 1,000 years (e.g., Model 1 in the NRC TPA code) (Mohanty, et al., 2002) as a surrogate for the effect of increased waste package temperature indicates a 150-percent increase in the peak expected dose relative to the nominal scenario (Figure 4-14).

<u>Uncertainties</u>

The following are key areas in which uncertainties exist:

- The effects of potential mechanical interactions between the drip-shield and waste package under rockfall and seismic conditions are uncertain.

- The effects of drift degradation on water seepage into the drifts are uncertain.

- The effect of elevated temperatures caused by backfill from drift degradation may have an important effect on the creep rate of the drip shield titanium alloys. The impact of the effects are uncertain at this time, however, analyses estimate that drift degradation is not expected to substantially increase the length of time that engineered barrier system components remain at significantly elevated temperatures.

- In the case of low-probability intrusive igneous activity event, magma would flow into the drifts because of the pressure gradient between the magma conduit and the drift opening (Woods, et al., 2002). An extensive blockage of the drifts by