RAI: Volume 3, Chapter 2.2.1.2.1, Fourth Set, Number 5:

Justify using average corrosion values for borated stainless steel (25-90ºC) and nickel-gadolinium alloy (30-60ºC) when the corrosion values range with temperature significantly and these temperatures are lower than repository conditions (see ANL–DS0–NU–000001).

1. RESPONSE

The information in this response provides justification that the corrosion rates used for the borated stainless steel and nickel-gadolinium (Ni-Gd) alloys are appropriate for their use in criticality analyses. The average corrosion rates reported in *Screening Analysis of Criticality Features, Events, and Processes for License Application* (SNL 2008a, Table 4.1-8) represent conservative values obtained on fully immersed specimens evaluated using short-term electrochemical and descaled weight loss testing techniques. In considering repository relevant temperatures, it is important to note that the maximum in-package temperature at the time of waste package breach will generally be relatively low, i.e., approximately 70°C or less.

For the commercial spent nuclear fuel (SNF) borated stainless steel alloy, the average corrosion rate value (27 nm/yr) was obtained using electrochemical techniques at 60°C. In the postclosure criticality analysis, the modeled corrosion rate for the borated stainless steel is 250 nm/yr, which is about 9 times the 27 nm/yr average measured corrosion rate for this material at 60°C. Also, the boron is contained in insoluble Cr_2B particles which, if removed from the absorber plates by corrosion, would likely still remain within the waste package between the fuel cell regions maintaining criticality control.

For the DOE SNF Ni-Gd alloy, average corrosion rates were obtained using accelerated electrochemical testing at 30°C and 60°C. Since the base composition of the Ni-Gd alloy is similar to Alloy C4, it is expected that the long-term corrosion rates for the Ni-Gd material under in-package chemistry conditions will approach the very low corrosion rates observed for Alloy C4. For the DOE SNF postclosure criticality analysis, the intact flooded canisters are subcritical even without the Ni-Gd neutron absorber and thus the absorber plate corrosion rate is immaterial for the intact condition. For the degraded condition, gadolinium alloy beads will be added to the DOE SNF canister as necessary to ensure that sufficient gadolinium is present in the degraded fuel mixture to demonstrate criticality safety (DOE 2004a, p. 87). Thus, even for the degraded condition, the postclosure criticality analyses is insensitive to Ni-Gd corrosion rates and no explicit corrosion rate was used. The identification of the specific bead form (i.e., material composition, density, size, gadolinium wt %, etc.) will be developed prior to acceptance of the various DOE SNF fuel types. As described in SAR Section 5.10.2.4.2 and Table 5.10-3, administrative controls will be compiled and maintained by the DOE in accordance with the requirements of the license specifications. SAR Section 2.2.1.4.1 presents the methodology and analyses required to confirm that waste forms are acceptable from a postclosure criticality perspective.

1.1 DESCRIPTION OF NEUTRON ABSORBER MATERIALS

The neutron absorber material used within the commercial SNF waste packages is borated stainless steel plate, while the DOE SNF waste packages use a Ni-Gd alloy in the form of plate (SNL 2008a, Sections 4.1.14 and 4.1.15; DOE 2008, Section 3.1.5). In the postclosure criticality analysis, the modeled corrosion rate for the borated stainless steel is 250 nm/yr, which is about 9 times the measured average corrosion rate for this material at 60°C (Table 1). This corrosion rate is reflected in SAR Section 1.5.1.1.1.2.2.2, which specifies an initial minimum absorber plate thickness of 11 mm and a minimum of 6 mm remaining after 10,000 years. The minimum thickness of 6 mm was used in developing the loading curve in *CSNF Loading Curve Sensitivity Analyses* (SNL 2008b, p. 7-2), as compared to the 10 mm implied by *Screening Analysis of Criticality Features, Events, and Processes for License Application* (SNL 2008a, p. 4-13). For the DOE SNF postclosure criticality analysis, the intact flooded canisters are subcritical even without the neutron absorber. The Ni-Gd alloy plates and gadolinium alloy beads as necessary are used as a mechanism to install a measurable quantity of gadolinium needed for postulated degraded conditions.

1.2 CORROSION BEHAVIOR OF NEUTRON ABSORBER MATERIALS

While the Engineered Barrier System (EBS) components and neutron absorber materials are designed to maintain their function in nominal repository environments over the first 10,000-year period after repository closure, degraded states are also considered (SNL 2008a). Corrosion rates for both DOE SNF and commercial SNF neutron absorber materials were obtained from measurements performed on specimens fully immersed in aqueous solutions simulating in-package relevant environments as well as more aggressive seepage environments. The bases and measured range of corrosion rates for each material are described below.

Table 1. Corrosion Rates of Waste Package Materials

Absorber Material	Corrosion Rate		Notes
Ni-Gd Alloy	Average Value	56 nm/yr	30°C, J-13 Solution (1)
Ni-Gd Alloy	Average Value	307 nm/yr	60°C, J-13 Solution (1)
Borated Stainless Steel – 400-hour Test ASTM 304B4 Grade A Alloy	Average Value	27 nm/yr	Range 60°C (2)
Stainless Steel Type 316 L	Median Value	3 nm/yr	30°C Fresh Water (3)

Source: (1) DOE 2004b.
(2) DTN: MO0803ECTBSSAR.001.
(3) DTN: MO0409SPAACRWP.000.

1.2.1 General Corrosion Rates for Borated Stainless Steel

The average corrosion rate value of 27 nm/yr for borated stainless steel reported in Table 1 was obtained from polarization resistance (PR) measurements on triplicate creviced specimens after 400 hours of exposure. The specimens were immersed at 60°C in an aerated solution predicted to be representative of in-package chemistry following a breach. No visual evidence of localized corrosion was observed on the specimens after immersion. Measured corrosion rates obtained

after chemically descaling and weighing the replicate specimens ranged from 42.3 to 95.6 nm/yr, with an average value of 62 nm/yr, similar to but somewhat higher than the PR measured rates. Because of the very short exposure times and low measured weight losses, test times did not meet the requirements of ASTM G-31 (ASTM G 31-72, Section 8.11.4) and thus were corroborative but were not considered appropriate for direct use. In the postclosure criticality analysis, the modeled corrosion rate for the borated stainless steel is 250 nm/yr, which is about 9 times the 27 nm/yr measured average corrosion rate for this material at 60°C (Table 1) or about 4.5 times the average value from the corroborative descaled weight-loss measurements.

1.2.2 General Corrosion Rates for Ni-Gd Neutron Absorber Material

As reported in Table 1, average corrosion rate values at 30°C and 60°C (56 nm/yr and 307 nm/yr, respectively) are listed for Ni-Gd plate material. These values were obtained from potentiostatic polarization test measurements on specimens exposed for 50 hours in a dilute groundwater type solution (J-13) and a more concentrated groundwater type solutions (50× J-13). The range of measured short-term corrosion rate values varied from 39 nm/yr to 67 nm/yr at 30ºC and from 172 nm/yr to 525 nm/yr at 60ºC (DOE 2004b, Table 17). The specimens were polarized to 0.2V (vs. SCE), a potential above the estimated open circuit potential (DOE 2004b, Section 4.4.2 and Tables 5 and 17). Therefore, these measured rates are higher than rates that would be obtained at open circuit potentials.

A longer term exposure test was also performed in the test solutions at 30°C. The measured corrosion rates tended to decrease with time with a reported rate after 6,720 hours exposure to J-13 solution of 20 nm/yr and a reported rate after 5,424 hours exposure to the more concentrated 50× J-13 solution of 89 nm/yr (DOE 2004b, Section 4.4 and Table 16). A subsequent continuation of these exposures resulted in measured corrosion rates of 8 nm/yr after just under 2 years exposure in the J-13 test solution at 30ºC (Mizia et al. 2006).

The base composition (without the gadolinium addition) of the Ni-Gd alloy is very similar to Alloy C4 (ASTM B 575-94). Therefore, the corrosion rates of the Ni-Gd alloy would be expected to approach corrosion rate values similar to Alloy C4 once the much less corrosion resistant gadolinide precipitates dissolve. The very low measured corrosion rates for Alloy C4 specimens exposed for one year at 60°C and 90°C in a concentrated, simulated acidified water (SAW) brine varied from about 20 to 50 nm/yr. Other long-term Alloy C4 corrosion test results, developed on the German repository program (Smailos 1993), show corrosion rates of weld deposited Alloy C4 after 12 months exposure in concentrated NaCl brines at 150°C to be about 70 nm/yr. Thus, it is expected that the long-term corrosion rates for the Ni-Gd material under in-package chemistry conditions will approach these very low measured Alloy C4 rates after the surface-breaking Ni_5Gd precipitates fully dissolve.

Because of the very low expected Ni-Gd corrosion rates in the range of highly dilute solution compositions predicted to form over time in a breached waste package (SNL 2007a, Section 6.3.1.3.1 and Table 6-3), the Ni-Gd plate material is expected to slowly thin but remain intact. However, the use of the Ni-Gd alloy is not a preventative approach to criticality for which ‘structure’ is credited. The various intact DOE SNF in a flooded canister are safe from a criticality standpoint whether the neutron absorber is present in the plates or not. During

degradation of waste packages containing DOE SNF, the gadolinium released during degradation of the basket material forms into $GdPO_4{:}2H_2O$ and $Gd_2(CO_3)_3$ (SNL 2007b, Figures E-12 and E-13), with no significant release of gadolinium (<1%) from the waste package (SNL 2007b, Section 8.1.1.1). Additional gadolinium will be added as needed in the form of gadolinium alloy beads to ensure sufficient gadolinium is present in the degraded fuel mixture.

1.3 EXPECTED MAXIMUM NEUTRON ABSORBER MATERIAL TEMPERATURE AT WASTE PACKAGE BREACH

As indicated in Figure 4 from *General Corrosion and Localized Corrosion of the Drip Shield* (SNL 2007c), the temperature range for the DOE SNF and the commercial SNF waste packages is about 60°C to 120°C after 1,000 years and about 30°C to 70°C after 10,000 years. Thus, in considering the effect of temperature on the corrosion rates of the neutron absorber materials, it is important to assess the estimated times (and associated temperatures) at which waste package breach can potentially occur as well as the type of breach (i.e., tight cracks versus gross failure of the outer corrosion barrier) for the various waste package outer corrosion barrier penetration mechanisms.

1.3.1 Gross Failure of the Waste Package Outer Barrier

In those cases where the breach is modeled as 'gross failure' (e.g., by early failure processes, igneous intrusion, seismic fault displacement, or localized corrosion), advective flow can occur, potentially resulting in more aggressive environments than for the case of tight cracks. The potential for gross failure and the impact of such failures on the expected waste package breach temperature is discussed below.

1.3.1.1 Drip Shield Early Failure from Emplacement Error

Drip shield early failure resulting from an emplacement error is treated along with other potentially relevant early failure mechanisms.. As discussed in SAR Section 2.4.1.2.2, as a bounding assumption, the waste package under an early-failed drip shield is assumed to experience localized corrosion over its entire surface as soon as seepage contacts the waste package. For criticality analyses, the probability of drip shield misplacement followed by localized corrosion is sufficiently low ($< 4.6 \times 10^{-5}$) to limit the contribution from this initiating event to an insignificant level (SNL 2008a, Section 6.3.2).

1.3.1.2 Igneous Intrusion

The effects of an igneous intrusive event on criticality are discussed in *Screening Analysis of Criticality Features, Events, and Processes for License Application* (SNL 2008a, Section 6.6.1). The initiating igneous intrusive event probability (1.7×10^{-4}) is a factor of 1,400 below the dominant contributor to the probability of criticality (i.e., seismic vibratory ground motion damaging the DOE SNF waste package (0.24)). Consequently, the probability of criticality resulting from the disruptive igneous scenario is sufficiently low that, if evaluated, would not change the conclusion that a criticality event in the repository can be screened from further consideration (SNL 2008a, Section 6.6.3).

1.3.1.3 Seismic Fault Displacement

Seismic events that can cause significant displacement (> 0.1 cm) along fault lines that intersect the drifts have a low probability of occurrence (i.e., mean annual exceedance frequencies of less than 10^{-6} per year) (SNL 2008a, Section 6.4). Additionally, new fractures that intersect the drift segments and the collapsing of the drift due to a seismic event will have an effect on the seepage location and rate. However, these changes in seepage have no impact on the repository's potential for criticality without drip shield failure (SNL 2008a, Section 6.4).

1.3.1.4 Localized Corrosion

Although advective flow can occur through regions removed by localized corrosion, as discussed in SAR Section 2.4.2.3.2.1.5, localized corrosion of the waste package outer corrosion barrier can only occur if seepage water with an aggressive chemistry contacts the waste package outer corrosion barrier surface (i.e., if the overlying drip shield is failed) within the first 12,000 years after closure. In the seismic ground motion modeling case, there is a low probability (3.4×10^{-4}) (SAR Section 2.4.3.2.2) of drip shield plate failure occurring before 10,000 years. As the probability of an improperly emplaced drip shield is also very low, localized corrosion of the waste package outer barrier in not expected during the 10,000 years following closure.

1.3.2 General Corrosion Openings in the Waste Package Outer Barrier

SAR Section 2.4.2.2.1.1.2 indicates general corrosion does not result in patch penetrations until about 500,000 years after repository closure. This is consistent with the discussion in excluded FEP 2.1.09.03.0B, Volume Increase of Corrosion Products Impacts Waste Package (SNL 2008c, FEP 2.1.09.03.0B), indicating that general corrosion, even under extreme exposure conditions (e.g., a constant exposure temperature of 150°C for 10,000 years), will not cause breach of the waste package outer corrosion barrier in the first 10,000 years after repository closure.

1.3.3 Cracks in the Waste Package Outer Barrier

With respect to stress corrosion cracking under nominal conditions, the waste package outer corrosion barrier closure lid region is low-plasticity burnished (stress remediated) to a minimum depth of 5 mm and stress corrosion crack growth rates are low such that, in the absence of larger weld flaws or seismic activity, cracks are not expected to penetrate the waste package outer corrosion barrier within 10,000 years of closure consistent with a neutron absorber material temperature of less than 70°C at the time of potential water ingress into the waste package. As discussed in excluded FEP 2.1.09.03.0B (SNL 2008c, FEP 2.1.09.03.0B), analyses estimate that about 16% of waste packages may have one or more weld flaws after ultrasonic inspection and repair. These remaining weld flaws will be generally small, with less than one percent being longer than the minimum stress-remediated depth of 5 mm. Larger weld flaws, e.g., those that extend beyond the stress remediated region, are relatively uncommon and must have an unfavorable combination of stress and stress intensity factor profiles and crack growth rates associated with them in order to penetrate the Alloy 22 waste package outer corrosion barrier within 10,000 years of closure.

Another potential source of cracks in the waste package outer barrier is stress corrosion cracking due to residual stresses generated during seismic activity. In order for seismic activity to result in crack penetrations on the waste package outer corrosion barrier within 10,000 years after closure, a seismic event with a mean annual exceedance probability of 4.3×10^{-4} or lower would have to occur (SNL 2008c, FEP 2.1.09.03.0B).

1.3.4 Expected Internal Environment Following Breach of Waste Package Outer Barrier

Stress corrosion cracks in passive alloys such as Alloy 22 tend to be tight and tortuous, often with many branches. Solution ingress through such tight cracks will be diffusive rather than advective (SNL 2008c, FEP 2.1.03.10.0A) and result in a much more benign waste package internal environment. Prior to drip shield breach, condensed water, with a low concentration of aggressive ions, could form on the underside of an intact drip shield (SNL 2008c, FEP 2.1.08.14.0A) and contact the waste package surface. Note that no credit is taken for the reduction in the rate of water ingress into a failed waste package due to the presence of the stainless steel inner vessel or, for commercial SNF, the transportation, aging, and disposal (TAD) canister, and for DOE SNF, the disposal canister. The interior of the waste packages will be warmer than the external environment for a considerable time period (SNL 2008d, Figure 6.4.2-4). The accumulation of water within the waste package will be limited by evaporation through the breaches because of the warmer waste package internal temperatures (SNL 2008e, Section 6.2.1.4). Because of vapor transport and drainage out of the waste package, the limited rate of liquid flow through cracks in the successive layers of the waste package will not be sufficient to cause significant ponding in the inner spent fuel canister in 10,000 years (SNL 2008e, Section 6.2.1.4). Thus, by the time sufficient water can enter the waste package under an early failed drip shield to result in flooding, it is likely the neutron absorber material will have cooled to well below 100°C.

Significant geometrical reconfigurations would be very improbable from waste package breaches that are limited to stress corrosion cracking since the internal configuration of the fuel relative to the absorber plate material will not significantly change. For the commercial SNF waste packages that use a borated stainless steel absorber material, the boron is contained in insoluble Cr_2B particles which, if removed from the absorber plates by corrosion, would likely still remain within the waste package between the fuel cell regions (SNL 2008a, p. 6-17) maintaining criticality control. For DOE SNF waste packages that use a Ni-Gd absorber material, the gadolinium released during degradation of the basket material forms into $GdPO_4{:}2H_2O$ and $Gd_2(CO_3)_3$ (SNL 2007b, Figures E-12 and E-13), with no significant release of gadolinium (<1%) from the waste package (SNL 2007b, Section 8.1.1.1).

The above considerations lead to the conclusion that by the time significant amounts of solution can contact the neutron absorber materials, there is a high likelihood that the absorber material temperatures will be relatively low (e.g., 70°C or below) and the resultant corrosion rates well within the conservative corrosion allowance margins specified.

1.4 SUMMARY AND CONCLUSIONS

The information provided in this response indicates that the corrosion rates used for the borated stainless steel and Ni-Gd alloys are appropriate for their use in criticality analyses. The average corrosion rates reported in *Screening Analysis of Criticality Features, Events, and Processes for License Application* (SNL 2008a, Table 4.1-8) represent conservative values obtained on fully immersed specimens evaluated using short-term electrochemical and descaled weight loss testing techniques. Maximum in-package temperatures at the time of waste package breach will generally be relatively low, i.e., approximately 70°C or less.

For the commercial SNF borated stainless steel alloy, the average corrosion rate value (27 nm/yr) was obtained using electrochemical techniques at 60°C. In the postclosure criticality analysis, the modeled corrosion rate for the borated stainless steel is 250 nm/yr, which is about 9 times the 27 nm/yr average measured corrosion rate for this material at 60°C. Also, the boron is contained in insoluble Cr_2B particles which, if removed from the absorber plates by corrosion, would likely still remain within the waste package between the fuel cell regions maintaining criticality control.

For the DOE SNF Ni-Gd alloy, average corrosion rates were obtained using accelerated electrochemical testing at 30°C and 60°C. Since the base composition of the Ni-Gd alloy is similar to Alloy C4, it is expected that the long-term corrosion rates for the Ni-Gd material under in-package chemistry conditions will approach the very low corrosion rates observed for Alloy C4. For the DOE SNF postclosure criticality analysis, the intact flooded canisters are subcritical even without the Ni-Gd neutron absorber, and thus the absorber plate corrosion rate is immaterial for the intact condition. For the degraded condition, gadolinium alloy beads will be added to the DOE SNF canister as necessary to ensure that sufficient gadolinium is present in the degraded fuel mixture to demonstrate criticality safety (DOE 2004a, p. 87). Thus, even for the degraded condition, the postclosure criticality analyses is insensitive to Ni-Gd corrosion rates and no explicit corrosion rate was used. The identification of the specific bead form (i.e., material composition, density, size, gadolinium wt %, etc.) will be developed prior to acceptance of the various DOE SNF fuel types. As described in SAR Section 5.10.2.4.2 and Table 5.10-3, administrative controls will be compiled and maintained by the DOE in accordance with the requirements of the license specifications. SAR Section 2.2.1.4.1 presents the methodology and analyses required to confirm that waste forms are acceptable from a postclosure criticality perspective.

2. COMMITMENTS TO NRC

None.

3. DESCRIPTION OF PROPOSED LA CHANGE

None.

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NOTE: [a] Provided as an enclosure to letter from Williams to Sulima dated 03/03/2009. "Yucca Mountain – Request for Additional Information Re: License Application, Safety Evaluation Report Volume 3, Chapters 2.2.1.2.1, Second Set - Submittal of U.S. Department of Energy Reference Citations."

RAI: Volume 3, Chapter 2.2.1.2.1, Fourth Set, Number 6:

Justify not taking into account localized corrosion of neutron absorber plates and oxide wedging effects of corrosion given that they were experimentally observed.

1. RESPONSE

As described in the response to RAI 3.2.2.1.2.1-4-005, the effects of oxidation/corrosion/ degradation have been considered in the development of the design basis configurations for both the DOE spent nuclear fuel (SNF) and commercial SNF waste forms and canisters. In considering the effects of localized corrosion and corrosion product oxide wedging effects on neutron absorber plate materials performance, it is important to distinguish between the two very different cases represented by DOE SNF and commercial SNF waste forms and canisters. For the DOE SNF waste forms, the neutron absorber structure is not necessary to prevent criticality, and thus localized corrosion or oxide wedging is of no consequence. For the commercial SNF, evaluations indicate that for the waste packages, the corrosion products produced through either general corrosion or localized corrosion act to displace moderator and have an overall negative reactivity effect for borated steel basket designs. Therefore, the current design basis configuration is a conservative representation for criticality analyses based on the transportation, aging, and disposal (TAD) canister specifications for the commercial SNF waste form, which use full density water in all void locations.

1.1 DOE SPENT NUCLEAR FUEL

1.1.1 Localized Corrosion Effects in DOE SNF

For the DOE SNF designs, the use of the Ni-Gd alloy plate material is not a preventative approach to criticality for which 'structure' is credited. The various intact DOE SNF, even in the unlikely event of a flooded canister, are safe from a criticality standpoint whether the absorber is present in the absorber plates or not. The absorber plates are used as a mechanism to install a measurable quantity of Gd needed for degraded conditions in postclosure. It is the degraded condition of the basket plates and potential for redistribution of the Gd throughout the degraded DOE SNF canister contents that is important to criticality safety for the time period when waste package breaching allows for moderation in and around the DOE SNF canisters. As described in more detail in the response to RAI 3.2.2.1.2.1-4-005, the nickel gadolinide intermetallic particles present in the Ni-Gd advanced neutron absorber plate material are much less corrosion resistant than the essentially Alloy C4 matrix (ASTM B 575-94) surrounding the particles. Consequently, as observed in corrosion studies of the Ni-Gd material, as the intermetallic gadolinide particles preferentially corrode and dissolve at the outer surface of the absorber plates (DOE 2004a, Section 4.4.1), the Gd remains within the waste package as discussed in Section 1.3.1. Simultaneously, as discussed in the response to RAI 3.2.2.1.2.1-4-005, the corroding, gadolinide-free surface of essentially Alloy C4 composition passivates, resulting in an extremely low effective corrosion rate. As described by Mizia et al. (2006, Section 5), "The gadolinide that intersects the surface exposed to acidic aqueous solutions and other solutions simulating the Yucca Mountain environment might be preferentially attacked and removed. However, the underlying Ni-Cr-Mo matrix will then repassivate, and the corrosion rate will drop off to an

extremely low rate." Consequently, since Alloy C4 is highly resistant to localized corrosion, the Ni-Gd plate material should also be highly resistant once the surface gadolinide particles are removed by selective corrosion and the remaining essentially Alloy C4 surface fully passivates.

1.1.2 Oxide Wedging Effects in DOE SNF Canisters

Oxide wedging is not expected to be a problem given the void volume (~50%) within most of the DOE SNF fuel canisters. Furthermore, the amount of 'rattle room' within each basket compartment (given the 'fuel footprint') will easily accommodate expansion of degraded corrosion products. Therefore, for the DOE SNF waste forms, the corrosion rate of the Ni-Gd material used is not relevant for maintaining criticality control, as the neutron absorbing material is only needed for certain degraded internal configurations. Therefore, these degraded configurations, which incorporate the various component corrosion products and internal reconfiguration to maximize system reactivity, are used to set the criticality control (neutron absorber) requirements. Thus, the effects from corrosion, corrosion product formation, and reconfiguration do not present a criticality potential.

1.2 COMMERCIAL SPENT NUCLEAR FUEL

1.2.1 Localized Corrosion Effects in Commercial SNF TAD Canisters

The initiation of localized corrosion requires the presence of liquid water, and therefore crevice corrosion of borated stainless steel neutron absorber plates cannot occur prior to breach of the waste package outer corrosion barrier. If a breach of the waste package outer barrier were to occur, liquid water could enter the breached package either in the form of condensation or seepage water. Ingress of seepage water cannot occur until the drip shield fails. Failure of the drip shield due to general corrosion is not predicted in the first 10,000 years and localized corrosion and stress corrosion cracking of the drip shield are excluded (SNL 2008a) by FEP 2.1.03.03.0B (Localized Corrosion of Drip Shields), FEP 2.1.09.28.0B (Localized Corrosion on Drip Shield Surfaces Due to Deliquescence), and FEP 2.1.03.02.0B (Stress Corrosion Cracking (SCC) of Drip Shields), respectively. Also, as discussed in the response to RAI 3.2.2.1.2.1-4-005, for the seismic ground motion modeling case, there is a low probability of drip shield plate material failing before 12,000 years after closure. Consequently, localized corrosion of the waste package outer barrier is not expected. Although there is a potential for ingress of condensation solution into the waste package following breach of the outer corrosion barrier, localized corrosion of the borated stainless steel due to condensation is not expected because it will be a benign environment with a low concentration of aggressive anions (e.g., chlorides) (SNL 2008a, FEP 2.1.08.14.0A, Condensation on Underside of Drip Shield). This lack of localized corrosion is consistent with the test results reported in the response to RAI 3.2.2.1.2.1-4-005. However, even for the cases where advection through the drip shield could occur followed by localized corrosion of the waste package outer barrier, the (conditional) probability that criticality would occur is found to not impact criticality screening (e.g., SNL 2008b, Sections 6.3.2 and 6.4.2.2). As described, the average measured corrosion rate (27 nm/yr) for borated stainless steel was obtained from polarization resistance (PR) measurements performed periodically over about a 575-hour exposure period on triplicate creviced specimens immersed at 60°C in a solution predicted to be representative of in-package

chemistry following a breach. The test solution ionic composition was 0.004 molal chloride, 0.001 molal fluoride, and 0.0025 molal nitrate with a pH of 5.50. At the end of the ~575-hour test period, a cyclic potentiodynamic polarization (CPP) test was performed on one of the specimens, and all specimens were then subjected to corroborative descaled weight loss measurement to establish an approximate corrosion rate. No visual evidence of localized corrosion was observed on the two specimens not subjected to the CPP test. The CPP-tested specimen had a staining around the edges of the crevice former, which was reported as likely due to a thin layer of corrosion product rather than to localized corrosion. No sign of gross free surface pitting or crevice damage was observed as a result of the test.

1.2.2 Oxide Wedging Effect in Commercial SNF Waste Packages

The effects of oxidation and oxide wedging, as well as the potential relevant configurations that can form, have been considered in the development of the design basis configuration. The design basis configuration is characterized by certain nonphysical representations in order to increase system reactivity and bound potential configurations that may occur over time.

The initial void volume within commercial SNF packages is less than the volume occupied if all the basket materials were to completely oxidize. For the commercial SNF, the corrosion products act to displace moderator and thereby reduce the effective neutron thermalization of the system. Various analyses have been performed during the evolution of the waste package design evaluating different modes of degradation and corrosion product buildup, including the effects of internal component oxidation and hydration. These evaluations ultimately indicate that for the commercial SNF waste packages the corrosion products act to displace moderator and have an overall negative reactivity effect for borated stainless steel basket designs. Therefore, the current design basis configuration is a conservative representation for criticality analyses based on the TAD canister specifications for the commercial SNF waste form, which use full density water in all void locations.

1.3 SUMMARY AND CONCLUSIONS

1.3.1 Localized Corrosion Considerations

Localized corrosion has not been observed on creviced borated stainless steel for environments relevant to commercial SNF in-package criticality. For the DOE SNF, the Ni-Gd material may undergo an initially high corrosion rate as the surface-breaking gadolinides corrode. However, regardless of corrosion rate, during degradation of waste packages containing DOE SNF, the Gd released during degradation of the basket material forms into $GdPO_4{:}2H_2O$ and $Gd_2(CO_3)_3$ (SNL 2007, Figures E-12 and E-13), with no significant release of Gd (<1%) from the waste package (SNL 2007, Section 8.1.1.1). Consequently, since the DOE SNF designs do not rely upon the Ni-Gd in a structural sense (DOE 2004b, p. 43), localized corrosion is of no consequence.

1.3.2 Oxide Wedging Considerations

For the DOE SNF fuel canisters, oxide wedging is not a limiting problem given the void volume (~50%) within most of the DOE SNF fuel canisters, along with the amount of 'rattle room' within each basket compartment (given the 'fuel footprint'). This internal volume will easily accommodate expansion of degraded fuel corrosion products. The neutron absorbing material is only needed for certain degraded internal configurations. These configurations incorporate the various component corrosion products and internal reconfiguration to maximize system reactivity, and are used to set the criticality control (neutron absorber) requirements. Thus, the effects from corrosion, including oxide wedging, do not present a criticality potential for the DOE SNF fuel.

For the commercial SNF waste packages, the initial internal void volume is less than the volume occupied if all the basket materials were to completely oxidize. For the commercial SNF, the corrosion products act to displace moderator and thereby reduce the effective neutron thermalization of the system. Various analyses have been performed during the evolution of the waste package design evaluating different modes of degradation and corrosion product buildup, including the effects of internal component oxidation and hydration. These evaluations ultimately indicate that for the commercial SNF waste packages the corrosion products act to displace moderator, resulting in an increase in assembly spacing, and have an overall negative reactivity effect for borated stainless steel basket designs. Therefore, the current design basis configuration is a conservative representation for criticality analyses based on the TAD canister specifications for the commercial SNF waste form, which use full density water in all void locations.

2. COMMITMENTS TO NRC

None

3. DESCRIPTION OF PROPOSED LA CHANGE

None

4. REFERENCES

ASTM A 887-89 (Reapproved 2004). 2004. *Standard Specification for Borated Stainless Steel Plate, Sheet, and Strip for Nuclear Application*. West Conshohocken, Pennsylvania: American Society for Testing and Materials.

ASTM B 575-94. 1994. *Standard Specification for Low-Carbon Nickel-Molybdenum-Chromium, Low-Carbon Nickel-Chromium-Molybdenum, and Low-Carbon Nickel-Chromium-Molybdenum-Tungsten Alloy Plate, Sheet, and Strip*. Philadelphia, Pennsylvania: American Society for Testing and Materials.

ASTM B 932-04. 2004. *Standard Specification for Low-Carbon Nickel-Chromium-Molybdenum-Gadolinium Alloy Plate, Sheet, and Strip*. West Conshohocken, Pennsylvania: American Society for Testing and Materials.

DOE (U.S. Department of Energy) 2004a. *Interim Report on the Corrosion Performance of a Neutron Absorbing Ni-Cr-Mo-Gd Alloy*. DOE/SNF/REP-086, Rev. 0. Idaho Falls, Idaho: U.S. Department of Energy, Idaho Operations Office. ACC: DOC.20040412.0001.

DOE 2004b. *Packaging Strategies for Criticality Safety for "Other" DOE Fuels in a Repository*. DOE/SNF/REP-090, Rev. 0. Idaho Falls, Idaho: U.S. Department of Energy, Idaho Operations Office. ACC: MOL.20040708.0386.

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SNL (Sandia National Laboratories) 2007. *Geochemistry Model Validation Report: Material Degradation and Release Model*. ANL-EBS-GS-000001 REV 02. Las Vegas, Nevada: Sandia National Laboratories. ACC: DOC.20070928.0010.

SNL 2008a. *Features, Events, and Processes for the Total System Performance Assessment: Analyses*. ANL-WIS-MD-000027 REV 00. Las Vegas, Nevada: Sandia National Laboratories. ACC: DOC.20080307.0003

SNL 2008b. *Screening Analysis of Criticality Features, Events, and Processes for License Application*. ANL-DS0-NU-000001 REV 00. Las Vegas, Nevada: Sandia National Laboratories. ACC: DOC.20080208.0001.

RAI: Volume 3, Chapter 2.2.1.2.1, Fourth Set, Number 10:

Provide information on CSNF burnup credit models validation including validation of isotopics depletion and criticality models, as referenced in the SAR. Justify use of the Δk_{ISO} value of negative 0.0249 in the criticality analysis. This bias aggregates positive and negative reactivity effects of all principal isotopes for RCA configurations while the actual net effect might be very different for the actual configurations, and no evaluation of these potential compensatory effects is provided.

1. RESPONSE

The methodology used to validate the SAS2H isotopic composition predictions is established in *Disposal Criticality Analysis Methodology Topical Report* (YMP 2003). The methodology is applied in *Isotopic Model for Commercial SNF Burnup Credit* (BSC 2004) to determine the bias and bias uncertainty in k_{eff} values for waste packages due to biases and uncertainties in the calculated spent fuel compositions. The bias and bias uncertainty value (negative 0.0249) documented in *Isotopic Model for Commercial SNF Burnup Credit* (BSC 2004) is based on the following:

(1) A comprehensive evaluation of available radiochemical assayed (RCA) spent nuclear fuel (SNF) samples (104 total samples: 74 pressurized water reactor (PWR) samples and 30 boiling water reactor (BWR) samples)

(2) Application of the direct difference method to propagate SNF isotopic composition biases and uncertainties to k_{eff} values for a representative waste package configuration

(3) Determination of bias and bias uncertainty using a non-parametric statistical method. In addition, 57 commercial reactor critical (CRC) state-points were evaluated and the validation study determined that the bias and uncertainty value from the evaluated RCA SNF samples is bounding as compared to the corresponding value from the evaluated CRC state-points (BSC 2004).

The evaluated RCA SNF samples are representative of the various types of assemblies and irradiation conditions that characterize the current spent fuel inventory, as shown in *Isotopic Model for Commercial SNF Burnup Credit* (BSC 2004, Table 7). The PWR RCA samples span an initial enrichment range of 2.453 wt % through 4.67 wt % ^{235}U enrichment, and burnup values from 6.92 GWd/MTU through 55.7 GWd/MTU. For comparison, the PWR loading curve (SAR Figure 2.2-7) spans an initial enrichment range of 2.0 wt % through 5.0 wt % ^{235}U enrichment, and burnup values from 3.8 GWd/MTU through 44.1 GWd/MTU. The BWR samples span an initial enrichment range of 2.53 wt % through 3.95 wt % ^{235}U enrichment, and burnup values from 2.16 GWd/MTU through 65.5 GWd/MTU. For comparison, the BWR loading curve (SAR Figure 2.2-8) spans an initial enrichment range of 4.5 wt % through 5.0 wt % ^{235}U enrichment, and burnup values from 4.3 GWd/MTU through 8.2 GWd/MTU. Note that validation is not required below the lower range of weight percent values on the loading curve (i.e., 2.0 wt % ^{235}U enrichment for PWR and 4.5 wt % ^{235}U enrichment for BWR)

because burnup credit is not taken in that region. Hence, the RCA samples provide good coverage of the initial enrichment and burnup ranges covered by the loading curves. The number of samples providing experimental data for each burnup-credit isotope is presented in *Isotopic Model for Commercial SNF Burnup Credit* (BSC 2004, Table 8).

The direct difference method is used to determine the bias and bias uncertainty in the waste package k_{eff} values as a result of bias and bias uncertainty in the calculated isotopic compositions for commercial SNF. Gauld (2003, Sections 5.1.5 and 6.1) demonstrates that the direct difference method produces similar bias and bias uncertainty values as other best estimate methods, such as the Monte Carlo uncertainty sampling and the sensitivity and uncertainty methods. The best estimate methods allow random uncertainties in the calculated nuclide concentrations to produce partially compensating reactivity effects (e.g., the reactivity effect of over-predicted concentration for a fission product may be partially cancelled out by the reactivity effect of under-predicted concentration for another fission product). A best estimate method is considered a more realistic method of estimation as compared to the "bounding method," which is considered to be highly conservative due to its use of isotopic composition uncertainties in such a way as to maximize their effects on system reactivity in a non-physical way (Gauld 2003). For each of the RCA SNF samples, the direct difference method requires two criticality calculations for the 21-PWR or the 44-BWR waste package, as applicable: one using the measured compositions and the other using the compositions calculated with SAS2H. Thus, the Δk differences between the k_{eff} values obtained as previously described directly determine the bias and bias uncertainty in the k_{eff} values caused by biases and uncertainties in the calculated isotopic compositions for the SNF samples.

Most of the 104 evaluated RCA SNF samples provide experimental data for the major actinides; however, 11 Three Mile Island (TMI) (4% ^{235}U initial enrichment and 44.8- to 55.7-GWd/MTU burnup values), 8 TMI (4.65 % ^{235}U initial enrichment and 22.8- to 29.9-GWd/MTU burnup values), three Calvert Cliffs (3.038 % ^{235}U initial enrichment and 44.8- to 55.7-GWd/MTU burnup values), and nine Mihama (3.203%, 3.208%, and 3.21% ^{235}U initial enrichment and 8.30- to 34.32-GWd/MTU burnup values) samples provide experimental data for 29, 22, 19, and 15 burnup-credit isotopes, respectively. Use of the direct difference method is justified since a significant number of samples were evaluated to test the SAS2H predictions for most of the burnup-credit isotopes over the range of burnup values between 8.30 and 55.7 GWd/MTU in a representative waste package configuration. The combined bias and bias uncertainty value (negative 0.0249) documented in SAR Section 2.2.1.4.1.1.2.4.1, which was obtained by applying the direct difference method, is comparable to combined bias and uncertainty values that have been determined elsewhere (see for example Gauld 2003, Sections 5.2 and 6.2). Note that some differences are expected because Gauld (2003) evaluated only 56 RCA samples and determined Δk_{ISO} values for a different configuration (a 32-PWR generic cask) by applying various propagation methods, including the direct difference method. Finally, note that additional non-credited conservatism is built into the design-basis configurations, as described in SAR Section 2.2.1.4.1.1.2.2, to account for uncertainties in the calculated isotopic compositions for parameters that may not be represented by the evaluated RCA SNF samples.

The distribution-free (i.e., nonparametric) tolerance limit method was determined to be applicable for determining Δk_{ISO} uncertainty since the Δk differences between the k_{eff} values obtained as previously described were not normally distributed (see the methodology established in YMP 2003, Section 3.5.3). This method involves sorting all Δk values in ascending order and determining the degree of confidence for the fraction of the true population that lies above the smallest observed values. Based on the nonparametric analysis, 95% of the Δk value population lies above the Δk_{ISO} value of negative 0.0249 with 95% confidence.

The main challenge associated with validation of the isotopic composition predictions is the availability and quality of measured SNF RCA data. Many years of successful reactor operations provide confidence in the ability of existing codes and data to accurately calculate reactivity associated with depletion and decay. From such experience, one would infer that the bias and bias uncertainty is much less than the 0.0249 value used in the SAR, which is driven largely by uncertainties in the RCA data (see the uncertainty and bias values in Gauld 2003, Sections 5.2 and 6.2). While such qualitative arguments are useful for perspective, they are not sufficient for quantitative validation, and hence the 0.0249 value is used in the SAR.

It is also worth noting that the isotopic validation approach used here is considerably more conservative than the standard approach used for burnup credit in spent fuel pools (Kopp 1998), which permits burnup credit for all nuclides. The guidance from Kopp (1998) is: "*A reactivity uncertainty due to uncertainty in the fuel depletion calculations should be developed and combined with other calculational uncertainties. In the absence of any other determination of the depletion uncertainty, an uncertainty equal to 5 percent of the reactivity decrement to the burnup of interest is an acceptable assumption.*" The latter portion of this guidance has become the standard approach for spent fuel pool criticality evaluations. Because the reactivity uncertainty is a percentage of the total reactivity decrease, the reactivity uncertainty value increases with burnup. Applying this approach to the 21-PWR waste package, in which the largest reactivity decrement corresponds to 5.0 wt % fuel burned to 44.1 GWd/MTU, the reactivity uncertainty value would be <0.012, which is less that half of the 0.0249 value used in the SAR. Given the small amount of burnup credit used in the 44-BWR waste package, the value would be considerably smaller in that case.

Quantitative validation performed in the referenced project documents, comparisons to independent work, comparison to guidance for burnup credit in spent fuel pools (under 10 CFR Part 50), and expectations based on reactor operating experience all provide support for the 0.0249 value as a reasonable, perhaps overly-conservative, value for accounting for biases and uncertainties in the isotopic composition predictions.

Validation of the criticality calculation methodology is documented in *Range of Applicability and Bias Determination for Postclosure Criticality of Commercial Spent Nuclear Fuel* (Radulescu et al. 2007). Refer to the responses to RAIs 3.2.2.1.2.1-4-028 through 3.2.2.1.2.1-4-033 for relevant details of the criticality model validation documented by Radulescu et al. (2007).

2. COMMITMENTS TO NRC

None.

3. DESCRIPTION OF PROPOSED LA CHANGE

None.

4. REFERENCES

BSC (Bechtel SAIC Company) 2004. *Isotopic Model for Commercial SNF Burnup Credit*. CAL-DSU-NU-000007 REV 00B. Las Vegas, Nevada: Bechtel SAIC Company. ACC: DOC.20041122.0003.

Gauld, I.C. 2003. *Strategies for Application of Isotopic Uncertainties in Burnup Credit*. NUREG/CR-6811 (ORNL/TM-2001/257). Prepared for the U.S. Nuclear Regulatory Commission by Oak Ridge National Laboratory, Oak Ridge, Tennessee, June 2003.

Kopp, L. 1998. "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants." Memorandum from L. Kopp (NRC) to T. Collins (NRC), August 19, 1998, with attachment. ACC: HQO.19990520.0004.

Radulescu, G.; Mueller, D.E.; Goluoglu, S.; Hollenbach, D.F.; and Fox, P.B. 2007. *Range of Applicability and Bias Determination for Postclosure Criticality of Commercial Spent Nuclear Fuel*. ORNL/TM-2007/127. Oak Ridge, Tennessee: Oak Ridge National Laboratory. ACC: LLR.20080324.0244.

YMP (Yucca Mountain Site Characterization Project) 2003. *Disposal Criticality Analysis Methodology Topical Report*. YMP/TR-004Q, Rev. 02. Las Vegas, Nevada: Yucca Mountain Site Characterization Office. ACC: DOC.20031110.0005.

RAI: Volume 3, Chapter 2.2.1.2.1, Fourth Set, Number 16:

Justify the use of the binomial distribution in calculating single and multiple misloads considering the existence of human performance dependency. The binominal distribution assumes that each assembly selection is independent from the previous selection. It does not consider the possibility that misloads may be related to one another for the same loading team.

1. RESPONSE

This RAI response pertains to obtaining reliability values for geologic repository operations area (GROA) operations during preclosure and/or offsite activities and processes. This is necessary in order for the current analysis basis for postclosure criticality screening to be satisfied. These processes include manufacturing, procedural compliance, documentation, and inspection. Because detailed manufacturing and waste loading operations have yet to be developed, preclosure and postclosure criticality analyses are based on reliability values calculated for substitute or surrogate processes involving generic processes, operations, and human error probabilities. Surrogate fabrication and operational processes (e.g., a material selection error in the waste package component fabrication process) with associated human error factors have been evaluated in the event trees in *Analysis of Mechanisms for Early Waste Package/ Drip Shield Failure* (SNL 2007). The use of generic reliability assessment values for evaluating operational and/or fabrication errors is expected to generate more limiting (i.e., higher probability of failure) results than are expected during actual fabrication and process operations performed by operators trained on those specific processes. The purpose of these analyses is to establish reasonable reliability limits and provide confidence that, once final manufacturing and loading operations have been established, the postclosure criticality analysis conclusions will remain valid. The DOE will implement appropriate administrative controls to ensure that the reliability limits used in the preclosure and postclosure safety analyses are met. The DOE will similarly rely on the quality assurance programs and other controls implemented by 10 CFR 71 and 72 certificate holders or licensees for control of manufacturing, fabrication, shipping, handling, storage and other related activities.

For the assembly loading process to be subject to human performance dependency, a common link among the selection and loading operations must exist. Currently, the actual loading process has not been defined, and therefore representative human error probabilities have been used based on a surrogate analysis documented in *Commercial Spent Nuclear Fuels Waste Package Misload Analysis* (BSC 2003). As the processes are developed, a detailed human factors analysis or equivalent will be performed in order to demonstrate that the reliability estimates in the SAR are achieved. The assumption of independence in loading operations is reasonable for the analysis of the surrogate analysis because additional independent controls and procedures will be implemented as determined necessary to eliminate the risk associated with human performance dependency in the loading operations and achieve the reliability values required. While the surrogate process employs a simplified binomial distribution to represent assembly selection, it takes no credit for potential administrative or engineered controls that would ensure the resulting reliability value can be achieved. Through the use of such augmented administrative or

engineered controls, the reliability value resulting from the analysis of the surrogate loading process can be achieved.

As discussed during the clarification call on 02/24/2009, SAR Section 5.10, *License Specifications*, provides a method of controlling the necessary limitations as conditions of the repository license. Specifically, SAR Table 5.10-3 describes a number of administrative controls to be included in the license specifications required by 10 CFR 63.43. Among the controls described in SAR Table 5.10-3 is the Waste Package Loading, Handling, and Emplacement Program, which will require the DOE to establish and implement a program to confirm that the loading, handling, and emplacement of waste packages will include provisions to ensure that waste packages are loaded, handled, and emplaced in accordance with the assumptions in the preclosure and postclosure safety analyses. This program will include those conditions or limitations upon repository operations necessary to provide assurance that the postclosure performance of the geologic repository will be consistent with the performance assessment, including the screening of postclosure criticality on the basis of low probability as described in SAR Section 2.2.1.4.1.

2. COMMITMENTS TO NRC

None.

3. DESCRIPTION OF PROPOSED LA CHANGE

None.

4. REFERENCES

BSC (Bechtel SAIC Company) 2003. *Commercial Spent Nuclear Fuel Waste Package Misload Analysis.* CAL-WHS-MD-000003 REV 00A. Las Vegas, Nevada: Bechtel SAIC Company. ACC: DOC.20031002.0005; DOC.20050728.0004.

SNL (Sandia National Laboratories) 2007. *Analysis of Mechanisms for Early Waste Package/Drip Shield Failure.* ANL-EBS-MD-000076 REV 00. Las Vegas, Nevada: Sandia National Laboratories. ACC: DOC.20070629.0002.

RAI: Volume 3, Chapter 2.2.1.2.1, Fourth Set, Number 17:

Justify that the end states of the following event sequences for 21-PWR ABS type waste packages are OK rather than POT-CRIT, as identified on page I-7 of CAL-WHS-MD-000003. The analysis of these event sequences failed to consider the effect of dependency (or dependence in THERP's terminology) on human performance. The analyses of other types of waste packages have the same issue.

- Sequences 12 to 20
- Sequences 23 to 31

1. RESPONSE

The probability of misloading a commercial spent nuclear fuel (SNF) assembly into a transportation, aging, and disposal (TAD) canister is used as part of the analysis that evaluates the low probability of a postclosure criticality. This misload probability was extracted from a realistic but generic misload analysis reported in *Commercial Spent Nuclear Fuel Waste Package Misload Analysis* (BSC 2003). In Section 1.1, DOE notes in a general statement that the justification requested pertains to preclosure/offsite operations that are yet to be fully defined and describes the administrative controls that DOE will impose to ensure that the repository performance will conform to assumptions of the performance assessment, including the screening of postclosure criticality, or require new analysis. In Section 1.2, DOE notes that the portion of the event tree that pertains to Sequences 12 to 20 and 23 to 31 does not apply to loading a TAD canister, and thus the probabilities evaluated in this portion of the event tree were not included in determining the probability of misloading a commercial SNF assembly into a TAD canister.

1.1 PRECLOSURE/OFF-SITE OPERATION AND POSTCLOSURE CRITICALITY

The justification requested by this RAI pertains to the probability of misloading an assembly into a TAD canister during preclosure and/or offsite processes in order to support the basis for the current analysis screening postclosure criticality from the performance assessment. These preclosure and/or offsite processes include manufacturing, waste loading operation, procedural compliance, documentation, and inspection. Because detailed manufacturing and waste loading operations have yet to be developed, the postclosure criticality analysis is based on probability values calculated for a generic analysis for misloading a package containing 21 assemblies from a pressurized water reactor (PWR) with absorber plates (21-PWR ABS). This surrogate analysis in combination with the probability of criticality given a waste form misload (SNL 2008a, Section 6.4, and further discussed in the response to RAI 3.2.2.1.2.1-4-022) establishes a reasonable reliability limit and provides confidence that, once manufacturing and loading operations have been established, the conclusions of the postclosure criticality analysis will remain valid. DOE will implement the necessary administrative controls during manufacturing and waste loading operations to ensure that the reliability limits used in the postclosure safety analyses are met.

As stated in the response to RAI 4.2.5.1-1-009, prior to acceptance of SNF, the DOE will review quality-affecting, work-related documentation of commercial licensees and certificate holders to ensure that the work has been performed in a manner that complies with applicable quality assurance requirements and, where applicable, the DOE Standard Contract. The DOE will not accept any material for which applicable quality assurance requirements have not been satisfied. Reviews that the DOE will perform and the circumstances under which such reviews will be performed will be identified in procedures to be developed in connection with a request by the DOE to receive and possess licensed materials.

As discussed during the public clarification meeting on 24 February 2009, SAR Section 5.10, *License Specifications*, provides a method of controlling the necessary limitations as conditions of the repository license. Specifically, SAR Table 5.10-3 describes a number of administrative controls to be included in the license specifications required by 10 CFR 63.43. Among the controls described in SAR Table 5.10-3 is the establishment and implementation of a Waste Package Loading, Handling, and Emplacement Program. This program will include those conditions or limitations upon waste package loading necessary for DOE to review to provide assurance that the postclosure performance of the geologic repository will be consistent with the performance assessment including the screening of postclosure criticality on the basis of low probability as described in SAR Section 2.2.1.4.1, specifically (SAR Table 5.10-3), by "adherence to the waste package loading procedure, the operator ensures that the waste form is placed in the appropriate waste package configuration to minimize the likelihood of criticality due to lack of neutron absorber, inappropriate geometric configuration, or excessive fissile material."

As described in SAR Section 5.10.2.4.2 and Table 5.10-3, the specific constraints on loading operations will be compiled in a Technical Requirements Manual, controlled by the DOE in accordance with the requirements of the license specifications. The Technical Requirements Manual will contain information necessary to support and implement the programs listed in the administrative controls section of the license specifications that are unique to the geologic repository, such as the Waste Package Loading, Handling, and Emplacement Program and the Waste Form and Waste Package Qualification Program, including controls needed to ensure reliability of manufacturing, procedural compliance, documentation, and inspection. These programs will delineate the waste receipt inspection and verification at the repository to confirm that the incoming waste form meets the waste acceptance criteria for criticality safety (SAR Table 5.10.3). SAR Section 2.2.1.4.1 presents the methodology and analyses required to confirm that waste forms are acceptable from a postclosure criticality perspective. The administrative controls described in SAR Section 5.10.2 will require that similar analyses be completed prior to receiving individual waste forms or waste package design configurations that are not explicitly analyzed in the license application.

As discussed in SAR Section 5.10.4, DOE will submit a proposed draft set of license specifications, including the specific constraints on loading operations, to the NRC prior to the NRC issuing a license to receive and possess SNF and high-level waste (HLW). The final set of license specifications issued by the NRC are expected to be incorporated as an appendix to the license to receive and possess SNF and HLW.

1.2 JUSTIFICATION OF CURRENT ANALYSIS BASIS

As noted in SAR Section 1.5.1.1, commercial SNF will be placed in TAD canisters, which are then placed in waste packages for disposal. The TAD canister is similar to a package containing 21 assemblies from a pressurized water reactor with absorber plates (SNL 2008b, Section 1.3) that was part of a generic analysis of misloading packages reported in *Commercial Spent Nuclear Fuel Waste Package Misload Analysis* (BSC 2003). In this RAI, the NRC has asked DOE to justify not including human dependency in Sequences 12 to 20 and Sequences 23 to 31 in calculating the probability of misloading 21-PWR ABS packages. However, Sequences 12 to 20 and Sequences 23 to 31 are not used to calculate the probability of misloading a 21-PWR ABS package. Sequences 12 to 20 refer to the operator *selecting* the wrong package (a 21-PWR package with control rods (CR) to load PWR fuel (BSC 2003, Table I-1)). Sequences 23 to 31 refer to the operator *requesting* the wrong package (a 21-PWR CR package) to load PWR fuel (BSC 2003, Table I-1). The 21-PWR CR package is not a TAD canister option in the license application, and so neither a *request* nor a *selection* error can be made for the TAD canister for 21-PWR assemblies.

Several facts are pertinent to evaluating the probability of misloading a commercial SNF assembly. First, three TAD canister designs are planned for commercial SNF: a 21-PWR design, a 12-PWR design, and a 44-BWR design (a package containing 44 assemblies from a boiling water reactor) (SNL 2008b, Section 1.3, Table 2.2-12). Second, the current BWR loading curve presented in *CSNF Loading Curve Sensitivity Analysis* (SNL 2008a, Figure 6.33) shows there is no potential for assembly misload. Finally, the discussion in *CSNF Loading Curve Sensitivity Analysis* (SNL 2008a, Section 6.4) states that only 2.1% of assemblies (1,990 PWR assemblies out of 93,770 PWR assemblies) could potentially be misloaded into a 21-PWR TAD canister.

Pertinent information on the possibility of misloading a TAD canister was available from a previous generic analysis (BSC 2003). The generic misload analysis is based on loading five different types of packages: 21-PWR ABS, 21-PWR CR, 12-PWR (for 1,955 long South Texas PWR assemblies), 44-BWR, and 24-BWR. The probability of misloading each of the five package types was reported separately, which allowed DOE to select an analogous configuration for the 21-PWR TAD canister. Because of the extra length of South Texas PWR assemblies, the probability of misloading the 12-PWR TAD was insignificant (BSC 2003, Figure I-1). The probability of potentially misloading 2.1% of the PWR assemblies into the 21-PWR ABS package, which is the same percentage of PWR assemblies that might be misloaded into the TAD 21-PWR canister, was reported as 1.18×10^{-5} per package (BSC 2003, Table 13). This probability value was used directly in the criticality screening analysis (SNL 2008b, Section 4.1.5). The probability value is derived from summing the probability from Sequences 3C and 10C of the event tree shown in Figure I-1 (BSC 2003, Table 13). Sequence 3C starts with the operator selecting the correct PWR assembly; however, the assembly has a record error (incorrectly identifying it as an assembly permissible to load) that is not detected by an independent inspector (BSC 2003, Table 27). Sequence 10C is the selection of an incorrect PWR assembly that is not detected by an independent inspector (BSC 2003, Table 27).

Once an off-site TAD loading operation is defined, DOE will require a demonstration that the combined probability of a commercial SNF assembly misload and resulting criticality (SNL 2008a) meets the reliability limit discussed in *Screening Analysis of Criticality Features, Events, and Processes for License Application* (SNL 2008b). However, as shown in the criticality screening analysis (SNL 2008b, Table 7.1-1) and further discussed in the response to RAI 3.2.2.1.2.1-4-007, misloading a DOE SNF co-disposal package coupled with seismic damage dominates the total probability of criticality. Specifically, the probability of misloading a commercial SNF TAD is only a minor contributor and could increase 2 orders of magnitude (all other factors remaining the same) before the probability of criticality from misloading a commercial SNF TAD coupled with seismic damage equaled the probability of criticality from misloading absorber material in a DOE SNF co-disposal package coupled with seismic damage.

1.3 CONCLUSIONS

The above discussion, along with responses to RAIs 3.2.2.1.2.1-4-012 through 3.2.2.1.2.1-4-016, provides assurance that the probability of misloading a CSNF assembly, as extracted from *Commercial Spent Nuclear Fuel Waste Package Misload Analysis* (BSC 2003), is reasonable and would be similar to results from analysis of an actual loading operation.

2. COMMITTMENTS TO NRC

None.

3. DESCRIPTION OF PROPOSED LA CHANGE

None.

4. REFERENCES

BSC (Bechtel SAIC Company) 2003. *Commercial Spent Nuclear Fuel Waste Package Misload Analysis.* CAL-WHS-MD-000003 REV 00A. Las Vegas, Nevada: Bechtel SAIC Company. ACC: DOC.20031002.0005.

SNL (Sandia National Laboratories) 2008a. *CSNF Loading Curve Sensitivity Analysis.* ANL-EBS-NU-000010 REV 00. Las Vegas, Nevada: Sandia National Laboratories. ACC: DOC.20080211.0001.

SNL 2008b. *Screening Analysis of Criticality Features, Events, and Processes for License Application.* ANL-DS0-NU-000001 REV 00. Las Vegas, Nevada: Sandia National Laboratories. ACC: DOC.20080208.0001.

RAI: Volume 3, Chapter 2.2.1.2.1, Fourth Set, Number 29:

UO_2 and mixed oxide (MOX) lattice critical experiments were combined based on a statistical test (Student's T-Distribution). This test is purely statistical and does not take into account differences in the physics of the two systems. Justify the combining of these data rather than analyzing the bias and uncertainty for these two systems separately.

1. RESPONSE

The criticality model validation is provided in *Criticality Model* (BSC 2004) and *Range of Applicability and Bias Determination for Postclosure Criticality of Commercial Spent Nuclear Fuel* (Radulescu et al. 2007), as noted in SAR Section 2.2.1.4.1.1.2.4.2. Radulescu et al. (2007) described the basis for combining applicable benchmark critical experiments for bias determination for commercial spent nuclear fuel (SNF). Critical experiments are combined based on their significant neutronic similarity to the commercial SNF in waste package configurations, as determined by the values of SCALE/TSUNAMI indices that quantify neutronic similarities between application and experimental systems, such as the integral index c_k. Integral index c_k values near unity identify systems that exhibit similar k_{eff} sensitivities to nuclides with high uncertainties in cross-section data (i.e., systems that will also exhibit similar calculational biases due to cross-section uncertainties). The bias and uncertainty of the UO_2 and mixed oxide (MOX) systems were analyzed separately. The validation documented by Radulescu et al. (2007) determined that the UO_2 lattice critical experiments are applicable only to waste packages containing fresh fuel, whereas the MOX lattice critical experiments were determined applicable only to waste packages containing irradiated fuel. Table 17 of Radulescu et al. (2007) provides a summary of the applicable critical experiments used for bias determination for the commercial SNF.

2. COMMITMENTS TO NRC

None.

3. DESCRIPTION OF PROPOSED LA CHANGE

None.

4. REFERENCES

BSC (Bechtel SAIC Company) 2004. *Criticality Model*. CAL-DS0-NU-000003 REV 00A. Las Vegas, Nevada: Bechtel SAIC Company. ACC: DOC.20040913.0008.

Radulescu, G.; Mueller, D.E.; Goluoglu, S.; Hollenbach, D.F.; and Fox, P.B. 2007. *Range of Applicability and Bias Determination for Postclosure Criticality of Commercial Spent Nuclear Fuel.* ORNL/TM-2007/127. Oak Ridge, Tennessee: Oak Ridge National Laboratory. ACC: LLR.20080324.0244.

RAI: Volume 3, Chapter 2.2.1.2.1, Fourth Set, Number 31:

Justify not performing bias trending as a function of different absorber or other structural materials. There is frequently a bias effect due to the use of different material cross sections.

1. RESPONSE

The criticality model validation for commercial spent nuclear fuel (SNF) is provided in *Range of Applicability and Bias Determination for Postclosure Criticality of Commercial Spent Nuclear Fuel* (Radulescu et al. 2007), as noted in SAR Section 2.2.1.4.1.1.2.4.2. The validation calculation documented by Radulescu et al. (2007) does not perform bias trending as a function of different absorber or other structural materials. The response to RAI 3.2.2.1.2.1-4-030 justifies using the selected predictor variables for the regression analyses documented by Radulescu et al. (2007).

2. COMMITMENTS TO NRC

None.

3. DESCRIPTION OF PROPOSED LA CHANGE

None.

4. REFERENCES

Radulescu, G.; Mueller, D.E.; Goluoglu, S.; Hollenbach, D.F.; and Fox, P.B. 2007. *Range of Applicability and Bias Determination for Postclosure Criticality of Commercial Spent Nuclear Fuel.* ORNL/TM-2007/127. Oak Ridge, Tennessee: Oak Ridge National Laboratory.
ACC: LLR.20080324.0244.

RAI: Volume 3, Chapter 2.2.1.2.1, Fourth Set, Number 33:

All the laboratory critical experiments involve fresh fuel isotopics. Justify why they are applicable to systems involving either newly-irradiated or significantly aged (over geological time frames) CSNF. Similarly, demonstrate why the commercial reactor criticals are applicable to systems with significantly aged CSNF.

1. RESPONSE

The criticality model validation for commercial spent nuclear fuel (SNF) is provided in *Range of Applicability and Bias Determination for Postclosure Criticality of Commercial Spent Nuclear Fuel* (Radulescu et al. 2007), as noted in SAR Section 2.2.1.4.1.1.2.4.2. The method employed by Radulescu et al. (2007) to identify applicable benchmark critical experiments and justify their use in bias determination is based on neutronic similarity between the design-basis waste package configuration for commercial SNF described in SAR Section 2.2.1.4.1.1.2.1 and benchmark critical experiments. The SNF compositions in all assemblies in the design-basis waste package configuration correspond to minimum acceptable burnup as a function of enrichment (Radulescu et al. 2007, Section 6.1), and hence are considerably more reactive than what will actually be loaded into the waste packages. Details of the cross-section sensitivity and uncertainty method employed by Radulescu et al. (2007) are provided in the responses to RAIs 3.2.2.1.2.1-4-026 and 3.2.2.1.2.1-4-029.

The critical experiments applicable to criticality model validation for commercial SNF include public laboratory critical experiments (LCE) involving fresh fuel and the Haut Taux de Combustion (HTC) LCEs (Radulescu et al. 2007, Section 6.4). The public LCEs include mixed-oxide (MOX) and low enriched uranium (LEU) experiments. Note that the plutonium-to-uranium ratio and the isotopic compositions of HTC fuel rod uranium and plutonium were designed to be similar to what would be found in a typical pressurized water reactor (PWR) fuel assembly that initially had an enrichment of 4.5 wt % ^{235}U and was burned to 37,500 MWd/MTU. The fuel material in those rods includes ^{234}U, ^{235}U, ^{236}U, ^{238}U, ^{238}Pu through ^{242}Pu, and ^{241}Am, which is present due to the decay of ^{241}Pu.

Based on SCALE/TSUNAMI cross-section sensitivity and uncertainty analyses, the benchmark experiments applicable to criticality model validation for commercial SNF, including newly irradiated fuel, are composed of approximately 220 MOX, HTC, and commercial reactor critical experiments (see the integral index c_k values in Radulescu et al. 2007, Appendix B) (i.e., no benchmark critical experiments with UO_2 fuel were identified as applicable for commercial SNF). The LEU benchmark critical experiments were determined applicable to criticality model validation only for commercial fresh fuel in waste package configurations (Radulescu et al. 2007, Table 6.4.2).

An examination of the SCALE/TSUNAMI-IP output files (Radulescu et al. 2007, Appendix F) indicates that the c_k covariance matrix is dominated by uncertainties in nuclear data for long-lived actinides such as ^{238}U, ^{235}U, and ^{239}Pu. Therefore, the values of bias and bias uncertainty determined based on critical experiments identified as applicable to the validation of

criticality calculations for design-basis configurations are considered bounding throughout the 10,000-year regulatory timeframe since significantly aged commercial SNF waste packages (with any significant reactivity) are thermal systems with SNF compositions dominated by the previously mentioned long-lived actinides. In other words, the set of critical experiments identified to be applicable to the design-basis configurations are expected to be largely the same as what would be identified for any of the various degraded conditions that exhibit significant reactivity, which can only be thermal systems. Analyses in *CSNF Loading Curve Sensitivity Analysis* (SNL 2008, Section 6.2.5), demonstrate that reactivity decreases as degradation increases, providing confidence in the bounding nature of the design-basis configurations and the appropriateness of basing the validation on the design-basis configurations.

2. COMMITMENTS TO NRC

None.

3. DESCRIPTION OF PROPOSED LA CHANGE

None.

4. REFERENCES

Radulescu, G.; Mueller, D.E.; Goluoglu, S.; Hollenbach, D.F.; and Fox, P.B. 2007. *Range of Applicability and Bias Determination for Postclosure Criticality of Commercial Spent Nuclear Fuel*. ORNL/TM-2007/127. Oak Ridge, Tennessee: Oak Ridge National Laboratory. ACC: LLR.20080324.0244.

SNL 2008. *CSNF Loading Curve Sensitivity Analysis*. ANL-EBS-NU-000010 REV 00. Las Vegas, Nevada: Sandia National Laboratories. ACC: DOC.20080211.0001.

RAI: Volume 3, Chapter 2.2.1.2.1, Fourth Set, Number 39:

> The current Topical Report (YMP/TR-004Q) states (p.1-5) that "The probability of one criticality occurring will be less than one over the entire repository for the first 10,000 years." The prior version of the topical report, for which the staff provided a Safety Evaluation Report (NRC, 2000, page 19), stated: "…a criticality frequency of 10^{-4} per year for all combinations of waste packages and waste forms." Are the two statements equivalent? If not, please address the differences and quantify the current version's probability number.

1. RESPONSE

Although the two statements referred to in this RAI are not precisely, mathematically equivalent as written, they were intended to be equivalent, as they were meant to serve the same purpose in the disposal criticality methodology approach. This criterion is used to guide decision processes for waste package design.

2. COMMITMENTS TO NRC

The DOE commits to update the license application as described in Section 3. The change will be included in a future license application update.

3. DESCRIPTION OF PROPOSED LA CHANGE

In SAR 2.2.1.4.1.1, p. 2.2-32, the text:

> A design probability criterion for the probability of a single criticality occurring to be less than one over the entire repository for the first 10,000 years is imposed as a criticality control limit. The total probability of criticality is compared to the design probability criterion (Item 6 of Figure 2.2-4), and, if equal to or greater than the design probability criterion, then criticality mitigating strategies are required that may include a redesign of the waste package, criticality controls, or other components as necessary to reduce the total probability of criticality and meet the design probability criterion. The design probability criterion has been met; therefore, the Item 6 decision point results in a "yes."

will be replaced with:

> The design probability criterion in Item 6 of Figure 2.2-4 represents a design decision process to help ensure that the probability of criticality remains sufficiently unlikely.

4. REFERENCES

None.

RAI: Volume 3, Chapter 2.2.1.2.1, Fourth Set, Number 40:

DOE has not justified its selection of zero for postclosure criticality administrative margin. SNF systems that have keff values significantly below 1.0 already have sufficient margin, and are not a concern. Provide the approximate proportion of the spent fuel inventory that exceeds a keff value (including uncertainties) of 0.98 (the minimum that has historically been considered acceptable for other NRC-regulated activities), for each of the nine types of DOE SNF, and CSNF. This should be done for both the intact and degraded fuel cases. For all those calculations that exceeded this keff value, provide a description of the system, as well as the initially estimated probability and calculated keff value. For configurations close to the keff limit that have degraded geometries, justify the margin, given potential uncertainty about the most reactive form and geometry.

1. RESPONSE

This response provides additional justification to what has been provided in the license application for the use of zero administrative margin for postclosure criticality evaluations. Different from deterministic regulations such as 10 CFR 70.61(d), which requires a demonstration that "under normal and credible abnormal conditions, all nuclear processes are subcritical, including use of an approved margin of subcriticality for safety", 10 CFR 63 does not require the prevention of criticality nor does it establish or require a criticality administrative margin (minimum margin of subcriticality) for the postclosure period. Setting this value to zero is consistent with the absence of the requirement in 10 CFR Part 63. Nevertheless, DOE criticality evaluations contain appropriate and considerable conservatisms for a risk-informed, performance-based approach, and adequately accounts for uncertainties such that an administrative margin is not needed.

Criticality calculations are used to determine the loading requirements (i.e., loading curves, linear fissile loading, amount of poison) to remain subcritical, and to support the determination of the probability of occurrence of criticality for comparison with proposed 10 CFR 63.342(a)[1].

Postclosure criticality evaluations are performed to determine what configurations yield the highest system k_{eff} values over the first 10,000 years after waste emplacement. Models were developed to bound potential relevant variations in materials, geometry, and neutron spectrum that occur as the internals of the spent nuclear fuel (SNF) canisters and waste packages change over long time periods. These evaluations were performed to determine the amount of neutron

[1] On March 13, 2009, the NRC final rule implementing revised standards of the U.S. Environmental Protection Agency was published in the Federal Register. See Implementation of a Dose Standard After 10,000 Years, 74 Fed. Reg. 10,811. The final rule is effective April 13, 2009 and does not contain any material differences from the proposed rule with respect to this response.

absorbing material, loading requirements, and/or basket configurations necessary to maintain subcriticality.

As described in SAR Section 2.2.1.4.1.1.2, to ensure the calculated k_{eff} value is always greater than the actual k_{eff} value, conservative assumptions and representations are used to define the design configurations for each SNF canister and waste form. Collectively, these assumptions and representations result in overestimation of the k_{eff} value and provide margin in the analysis predictions and loading curves. Bias and bias uncertainty are accounted for in establishing the point where the system is considered to be potentially critical (i.e., critical limit) as presented in SAR Table 2.2-11.

As discussed and illustrated in the response to RAIs 3.2.2.1.2.1-4-34 and 3.2.2.1.2.1-4-35, critical limits were established for each of the nine representative DOE SNF criticality groups based on maintaining subcriticality for the most reactive, credible degraded configurations. The criticality limits are lower than the lower bound tolerance limits accounting for bias and bias uncertainty. As a result, additional design margin is included for the DOE SNF waste forms.

1.1 SPECIFIC RAI QUESTIONS

Although DOE does not believe that 10 CFR Part 63 requires any specific subcriticality margin, and as such is unaware of any regulatory requirement to apply a k_{eff} value of 0.98 for SNF, it is the Department's intent to be responsive to the NRC's RAI. The information below is provided as a direct response to the questions in the RAI and to provide a better understanding of how a significant amount of uncredited margin is incorporated into the design basis representations.

1. *Provide the approximate proportion of the spent fuel inventory that exceeds a k_{eff} value (including uncertainties) of 0.98 (the minimum that has historically been considered acceptable for other NRC-regulated activities), for each of the nine types of DOE SNF, and CSNF.*

Changes in the critical limit have a potential to affect the number of pressurized water reactor (PWR) SNF assemblies that are acceptable for loading per the loading curve, as described in SAR Section 2.2.1.4.1.3.2 (SAR, p. 2.2-50). Sensitivity evaluations are provided in *CSNF Loading Curve Sensitivity Analysis* (SNL 2008a, Section 6.4) for a misload analysis for the PWR waste package. The results presented in *CSNF Loading Curve Sensitivity Analysis* (SNL 2008a, Tables 6-42 to 6-47) show that assemblies with burnup/enrichment combinations resulting in the highest k_{eff} values, comprising 36 out of 1,990 assemblies on the unacceptable side of the loading curve, are the only assemblies for which a misload results in a violation of the current critical limit. A stochastic analysis was performed randomly loading the waste package. As stated in *CSNF Loading Curve Sensitivity Analysis* (SNL 2008a, p. 6-91), the combined failure (i.e., calculated k_{eff} exceeds critical limit) probability, considering all potential misload locations within the basket was 0.2%. To be conservative and provide additional margin in the criticality screening probability calculation, the value corresponding to the most reactive basket location (center position) was used (i.e., 1.4%) (see SAR Section 2.2.1.4.1.3.2, SAR p. 2.2-50).

The impact on the fraction of commercial SNF PWR waste form that is non-loadable based on imposing an additional 0.02 margin in k_{eff} is evaluated by considering information provided in Table 1.

Table 1. Non-Loadable* PWR Fuel Inventory

Burned Fuel Critical Limit	Number of Assemblies	% Inventory
0.9300	4,860	5.2
0.9400	3,289	3.5
0.9500	2,165	2.3
0.9529	1,990	2.1
0.9600	1,534	1.6
0.9700	1,083	1.2
0.9800	857	0.9
0.9900	792	0.8
1.0000	698	0.7

Source: SNL 2008a, Table 6-29.

*Non-loadable means not loadable without additional criticality control measures.

As shown in Table 1, changing the critical limit from 0.9529 to 0.9329 results in less than 5.2% of the total commercial SNF PWR inventory having the potential of exceeding a k_{eff} of 0.98. This is an increase in the non-loadable fraction of assemblies of less than 3.1%.

Based on the current loading curve, the design basis waste stream inventory, and using the design basis configuration, there are fewer than 4,860 PWR SNF fuel assemblies that exceed a k_{eff} (including uncertainties) of 0.98.

The analysis of commercial SNF boiling water reactor (BWR) assemblies is not sensitive to changes in the critical limit. The analysis has sufficient margin such that decreasing the critical limit for this waste form up to 0.05 will not result in a change to the loading curve significant enough to result in any BWR assemblies from the representative inventory being on the unacceptable side of the loading curve.

For DOE SNF, the proportion of the spent fuel inventory that exceeds a k_{eff} of 0.98 (including uncertainties) is 0 by canister loading requirements and poison system design. The canister loading requirements (e.g., fissile mass loading, poison requirements) account for bias and uncertainty in the calculations and the criticality controls will be set to ensure that the most reactive credible configuration remains below the critical limit including any additional margin (see the response to RAI 3.2.2.1.2.1-4-035 for estimated margins).

Therefore, the proportion of the spent fuel inventory that exceeds a k_{eff} value (including uncertainties) of 0.98 is approximately 2% based on using the number commercial SNF assemblies and the total number of DOE SNF canisters (some DOE SNF canisters contain incomplete fuel assemblies and/or individual fuels rods).

2. *This should be done for both the intact and degraded fuel cases.*

The information provided in Question 1 describes the design basis configuration which represents the most reactive credible configuration (intact for commercial SNF or degraded for DOE SNF). Including similar information for configurations that are less reactive than the design basis configuration is not necessary as most configurations have k_{eff} values that are significantly below 1.0 and as stated in the RAI, "SNF systems that have k_{eff} values significantly below 1.0 already have sufficient margin, and are not a concern." In addition, the licensing basis (and screening justification) is based on the design basis configuration.

3. *For all those calculations that exceeded this k_{eff} value, provide a description of the system, as well as the initially estimated probability and calculated k_{eff} value.*

The design basis configuration has been developed to incorporate certain nonphysical combinations of parameters to develop a bounding configuration that accounts for various changes in materials and geometry that can occur over time. The design basis configuration is used to define and establish the loading requirements.

For commercial SNF, the intact fuel configurations have the maximum reactivity. Using the current critical limit and loading curve, the maximum k_{eff} value is less than 0.9529 for burned fuel and less than 0.9905 for fresh fuel. The probability of criticality for this configuration is 0 when waste packages are loaded with transportation, aging, and disposal (TAD) canisters in accordance with design specifications, and in the event of a design nonconformance (i.e., component manufacturing error or assembly loading error), the probability is 6.4×10^{-7} for the entire repository over 10,000 years (SNL 2008b, Table 7.1-1).

For DOE SNF, the degraded configurations result in the maximum reactivity. The maximum reactivity value is criticality group specific as listed in RAI response 3.2.2.1.2.1-4-035, but will be 0.92, 0.93, or 0.97. The probability of criticality for this configuration is 0 when waste packages are loaded with DOE SNF canisters in accordance with design specifications, and in the event of a design nonconformance (i.e., exceeding fissile mass loading or insufficient amount of poison added) the probability is 3.6×10^{-5} for the entire repository over 10,000 years (SNL 2008b, Table 7.1-1 (calculated by subtracting 6.4×10^{-7} from 3.7×10^{-5})).

4. *For configurations close to the k_{eff} limit that have degraded geometries, justify the margin, given potential uncertainty about the most reactive form and geometry.*

Commercial SNF modeling simplifications are performed that result in higher k_{eff} values to bound potential relevant variations in materials, geometry, and neutron spectrum that occur as the internals of the waste packages change with time. The unquantified margin is included in the conservatisms imposed on the design basis configuration. Even with the inclusion of these conservatisms, the degraded commercial SNF configurations are less reactive than the intact configuration.

An example of conservative modeling parameters that are inherent in the design basis configuration are the use of the 6-mm-thick neutron absorber plates with credit for only 75% of

the boron content. The TAD canister design specifications require absorber material 0.4375-in. (11.1125-mm) thick when fabricated. Based on corrosion test data the expected mean corrosion rates of the borated stainless steel are 0.0271 μm/yr (SNL 2008b, Table 4.1-8). After 10,000 years of simulated corrosion attack on both faces of the plate, this would result in greater than 10mm of absorber plate remaining. Sensitivity cases were evaluated for different thicknesses of absorber plate in *CSNF Loading Curve Sensitivity Analysis* (SNL 2008a, Table 6-19) and show an average Δk_{eff} of 0.016 with plate thicknesses between 6 and 9 mm. Another example of conservatism is the use of the 5-year decayed isotopic compositions for commercial SNF. Not only are the SNF material compositions based on conservative depletion parameters that maximize end-of-life reactivity, no credit is taken for the decay beyond 5 years. Based on the repository thermal output, water (as assumed in these design basis criticality calculations) will not enter the repository for hundreds of years. The conservatism of the 5-year decay time compositions to be used to represent spent fuel compositions for the postclosure period has been demonstrated in *Principal Isotope Selection Calculation* (CRWMS M&O 1998, Section 6.1) and *Recommendations on the Credit for Cooling Time in PWR Burnup Credit Analyses* (Wagner and Parks 2003, Section 2.2) and can range from less than 1% to over 6% depending on the decay time, enrichment, and burnup. Other conservative modeling assumptions are listed in SAR Section 2.2.1.4.1.1.2.2 (pp. 2.2-33 to 2.2.36).

For DOE SNF canisters, the degraded configuration is more reactive than the nominal configuration. As with commercial SNF, conservative assumptions and representations are used to define the design configurations in performing the criticality calculations for each configuration. Collectively, these assumptions and representations result in overestimation of the k_{eff} value and provide margin in the analysis predictions. Also, DOE SNF criticality limits are lower than the lower bound tolerance limits accounting for bias and bias uncertainty. As a result, additional design margin up to Δk_{eff} of 0.0572 is included (see response to RAI: 3.2.2.1.2.1-4-035) in the criticality evaluations.

2. COMMITMENTS TO NRC

The DOE commits to update the license application as described in Section 3. The change will be included in a future license application update.

3. DESCRIPTION OF PROPOSED LA CHANGE

A typographical error was found in the critical limit values for rows one and three for the waste form, commercial SNF in SAR Table 2.2-11. This critical limit is for fresh fuel configurations and the isotopic bias and uncertainty term were inappropriately subtracted from it in the table. This typographical error does not impact any criticality evaluation or result as it is only an illustrative error.

Rows 1 and 3 of Table 2.2-11:

$$0.9905\ (0.0977 \leq \text{EALF} \leq 0.3882\ \text{eV})\ \text{Fresh Fuel}^{a} - 0.0249 = 0.9656$$

will be replaced with:

0.9905 (0.0977 ≤ EALF ≤ 0.3882 eV) Fresh Fuel[a].

4. REFERENCES

CRWMS M&O, 1998. *Principal Isotope Selection Calculation*. B00000000-01717-0210-00031 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19980806.0608.

SNL (Sandia National Laboratories) 2008a. *CSNF Loading Curve Sensitivity Analysis*. ANL-EBS-NU-000010 REV 00. Las Vegas, Nevada: Sandia National Laboratories. ACC: DOC.20080211.0001.

SNL (Sandia National Laboratories) 2008b. *Screening Analysis of Criticality Features, Events, and Processes for License Application*. ANL-DS0-NU-000001 REV 00. Las Vegas, Nevada: Sandia National Laboratories. ACC: DOC.20080208.0001; DOC.20080317.0008; LLR.20080401.0255; LLR.20080423.0161.

Wagner, J.C. and Parks, C.V. 2003. *Recommendations on the Credit for Cooling Time in PWR Burnup Credit Analyses, NUREG/CR-6781* (ORNL/TM-2001/272), prepared for the U.S. Nuclear Regulatory Commission by Oak Ridge National Laboratory, Oak Ridge, Tennessee.