RAI: Volume 3, Chapter 2.2.1.2.1, Fourth Set, Number 1:

Justify using the number of waste packages rather than the number of neutron absorber plates, tubes or basket components to calculate the probability of criticality for TAD and DOE canisters. This is based on a neutron absorber selection error and the neutron absorbers are individual plates, tubes or basket components that may not be manufactured from the same lot of material (see ANL–EBS–MD–000076 Rev 00).

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. 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.

The neutron absorber misload event as used in the probability evaluations represents the improper performance of the neutron absorber plates due to fabrication-related errors. Although this event is referred to as an absorber misload event, it actually refers to basket fabrication errors in general. Errors in fabrication and operational processes are primarily due to human factors. Since the detailed manufacturing and installation processes of the baskets are still to be developed, a surrogate analysis based on material selection errors was used. In this surrogate analysis, the occurrence of such errors is considered per waste package rather than per component. The individual subprocesses related to manufacturing the materials, installation of the absorber components, and ensuring that the basket meets the reliability estimates used in the probability calculations are discussed in the responses to RAIs 3.2.2.1.2.1-4-002 and 3.2.2.1.2.1-4-003.

Additionally, because the manufacturing and installation of the neutron absorber will be accomplished in accordance with standard nuclear industry practices, requirements, and procedures (e.g., *ASME Boiler and Pressure Vessel Code* (ASME 2001, Section III, Division 1, Subsection NC) for Class 2 components) that are used for storage and transportation casks, the manufacturer/supplier and/or installer will be required to perform independent inspections. After manufacturing, testing will be conducted using an adequate inspection method (e.g., wet chemistry and/or neutron attenuation techniques) to verify that the minimum neutron absorbing material specifications have been met. These tests will be performed using written and approved procedures, with results documented as part of the quality records documentation package. As discussed in the response to RAI 3.2.2.1.2.1-4-004, human failure to operate test equipment correctly is the dominant contributor to failure of the testing process. Because testing procedures and equipment are still to be specified, it is reasonable to consider failure to operate test equipment correctly as one event per testing cycle. As part of waste acceptance, the Yucca Mountain Project will inspect the material certifications and the fabrication records upon receipt by using existing neutron absorber testing methods that are currently accepted by the NRC to meet the transportation requirements of 10 CFR Part 71.

In summary, the probability of criticality due to neutron absorber selection error is appropriately based on the number of waste packages.

2. COMMITMENTS TO NRC

None.

3. DESCRIPTION OF PROPOSED LA CHANGE

None.

4. REFERENCES

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 2:

Justify how additional independent field material inspections will be implemented to ensure reduction of material selection error in absorber plate/tube/basket manufacturing from 1.0×10^{-4} down to 1.25×10^{-7}.

1. RESPONSE

This response provides additional justification that the reduction of material selection error in absorber plate/tube/basket from 1.0×10^{-4} down to 1.25×10^{-7} is reasonable by implementing additional inspections.

This RAI response pertains to obtaining reliability values for preclosure geologic repository operational area (GROA) and off-site operations and processes. These processes include independent field material inspections. Because inspection procedures have yet to be developed, postclosure criticality analyses are based on reliability values calculated for surrogate processes involving generic inspection methods. Surrogate inspection processes (e.g., confirmation of selecting the proper material through detection) have been evaluated in *Analysis of Mechanisms for Early Waste Package/Drip Shield Failure* (SNL 2007). Analyses based on these surrogate processes establish reasonable reliability limits and provide confidence that, once final inspection procedures have been established, the postclosure criticality analysis conclusions will remain valid. The DOE will implement appropriate administrative controls with sufficient reliability that the limits used in the postclosure criticality analysis are met.

Material field testing employs sufficient independence to reasonably justify a reduction in the probability of a material selection error from 1.0×10^{-4} down to 1.25×10^{-7}. As stated in *Analysis of Mechanisms for Early Waste Package/Drip Shield Failure* (SNL 2007, p. 6-37), whether the technology exists or a set of procedures is established to ensure the selection of the correct material, there is still the unlikely possibility that the technician in charge could fail to select the correct material, with a human error probability (HEP) of 1.25×10^{-3}. This HEP is within the range documented by Hollnagel (1998, Chapter 9, Table 9, p. 252) for similar tasks (e.g., wrong object observed). Given that the verification of the neutron absorber plate in a transportation, aging, and disposal (TAD) canister is conducted independently at a waste custodian site or in a GROA facility (away from the TAD vendor facilities), and performed by a different group of operators, the HEP represents a failure independent of the failure to fabricate the neutron absorber. In other words, the value of 1.25×10^{-7} represents the product of the probability of two independent errors: the probability that an incorrect material is selected (1.0×10^{-4}) times the probability that this error is not noticed in an independent verification (1.25×10^{-3}).

As stated in the response to RAI 4.2.5.1-1-009, prior to acceptance of commercial SNF, the DOE will perform reviews of commercial licensees and certificate holder quality-affecting work-related documentation 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 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. As described in SAR Section 5.10.2.4.2 and Table 5.10-3, these administrative controls will be compiled in the Technical Requirements Manual, 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 and canisters are acceptable from a postclosure criticality perspective. The administrative controls described in SAR Section 5.10.2 and SAR Table 5.10-3 require that similar analyses be completed prior to receiving individual waste forms or canisters/waste package design configurations that are not explicitly analyzed in the license application.

2. COMMITMENTS TO NRC

None.

3. DESCRIPTION OF PROPOSED LA CHANGE

None.

4. REFERENCES

Hollnagel, E. 1998. *Cognitive Reliability and Error Analysis Method, CREAM*. 1st Edition. New York, New York: Elsevier.

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 3:

Justify application of material selection error in absorber plate/tube/basket manufacturing of 1.0×10^{-4} using pressure vessel manufacturer's data while neutron absorber plates/tubes/baskets are not required to be manufactured to ASME Boiler and Pressure Vessel Code (Class 1 or Class 2 components) as waste package and TAD canister (see ANL-EBS-MD-000076 Rev 00).

1. RESPONSE

This response provides additional justification that applying material selection error from pressure vessel manufacturer's data to neutron absorber plates is reasonable.

This RAI response pertains to obtaining reliability values for off-site operations and processes associated with neutron absorber plate/tube manufacturing. These processes include material selection error during manufacturing. Because manufacturing processes have yet to be developed, postclosure criticality analyses are based on reliability values calculated for surrogate processes involving generic manufacturing methods. Surrogate manufacturing processes (e.g., those associated with ASME Boiler and Pressure Vessel Code) have been evaluated in *Analysis of Mechanisms for Early Waste Package/Drip Shield Failure* (SNL 2007). Analyses based on these surrogate processes establish reasonable reliability limits and provide confidence that, once final inspection procedures have been established, the postclosure criticality analysis conclusions will remain valid. The DOE will implement appropriate administrative controls with sufficient reliability that the limits used in the postclosure criticality analysis are met.

Office of Civilian Radioactive Waste Management (OCRWM) Transportation, Aging and Disposal (TAD) Canister System Performance Specification (DOE 2008) includes manufacturing requirements that must comply with ASTM A887-89 for neutron absorber plates or tubes and ASTM-A-276-06 for structural internals (baskets). These standards contain stringent and thorough material chemical and tensile requirements; they also specify requirements for verification and certification.

ASTM A887-89, *Standard Specification for Borated Stainless Steel Plate, Sheet, and Strip for Nuclear Application* (2004, Section 8.4) specifies that boron determination not covered in ASTM A751, *Test Methods, Practices and Terminology* shall be made as agreed upon between the purchaser and supplier. Hence, testing and examination that can detect light material such as boron in stainless steel plates will be specified for material selection verification.

This application of the ASTM standards will result in process controls for neutron absorber manufacturing that are similar to and as robust as the ASME codes for the specific task of validating proper material selection. Therefore, the error data for the material selection process in pressure vessel manufacturing that is used in the postclosure criticality analysis can be utilized as a material selection error for absorber plate/tube manufacturing.

The TAD specification and ASTM A887-89 requirements for the neutron absorber plates/tubes provide significant barriers to material specification errors. In addition, ASTM codes

specifications provide supplementary inspection types and/or requirements that enable the supplier and purchaser to agree to additional special testing as needed. Similar supplementary inspection types and/or requirements will be specified to ensure the postclosure criticality analysis basis of material selection error in absorber plate/tube manufacturing of 1.0×10^{-4} is met.

As stated in the response to RAI 4.2.5.1-1-009, prior to acceptance of commercial spent nuclear fuel, the DOE will perform reviews of commercial licensees and certificate holder quality-affecting work-related documentation 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 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. As described in SAR Section 5.10.2.4.2 and Table 5.10-3, these administrative controls will be compiled in the Technical Requirements Manual, 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 and canisters are acceptable from a postclosure criticality perspective. The administrative controls described in SAR Section 5.10.2 and SAR Table 5.10-3 require that similar analyses be completed prior to receiving individual waste forms or canisters/waste package design configurations that are not explicitly analyzed in the license application.

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.

DOE (U.S. Department of Energy) 2008. *Transportation, Aging and Disposal Canister System Performance Specification.* WMO-TADCS-000001, Rev. 1 ICN 1. Washington, D.C.: U.S. Department of Energy, Office of Civilian Radioactive Waste Management.
ACC: DOC.20080331.0001.

RAI: Volume 3, Chapter 2.2.1.2.1, Fourth Set, Number 4:

Justify that portable X-ray inspection equipment can reduce the material selection error of neutron absorbers by at least three orders of magnitude since the X-ray equipment described cannot measure light elements such as boron (see ANL-EBS-MD-000076 Rev 00).

1. RESPONSE

This response provides additional justification that the use of reliability data associated with x-ray detection equipment to reduce the material selection error of neutron absorbers is reasonable even though x-ray equipment cannot detect boron.

This RAI response pertains to obtaining reliability values for preclosure geologic repository operational area (GROA) and off-site operations and processes. These processes include independent field material inspections. Because inspection procedures have yet to be developed, postclosure criticality analyses are based on reliability values calculated for surrogate processes involving generic inspection methods. Surrogate inspection processes (e.g., confirmation of selecting the proper material through x-ray detection) have been evaluated in *Analysis of Mechanisms for Early Waste Package/ Drip Shield Failure* (SNL 2007). Analyses based on these surrogate processes establish reasonable reliability limits and provide confidence that, once final inspection procedures have been established, the postclosure criticality analysis conclusions will remain valid. The DOE will implement appropriate administrative controls with sufficient reliability that the limits used in the postclosure criticality analysis are met.

Portable x-ray spectroscopy is used as a surrogate inspection method for the purpose of reliability estimation of potential instrumentation that may be utilized in the field for measurements of material composition (SNL 2007, Section 6.3.2). It is recognized that equipment such as portable thermal neutron radiography can detect light material such as boron in steel plate and could be specified in the future. Such a portable instrument will be available at the waste custodian sites or GROA facilities for the confirmation that neutron absorber plates or tubes are indeed installed in the transportation, aging, and disposal (TAD) canisters prior to loading fuel assemblies. Failure of such a confirmation task can come from two potential sources: instrumentation failure or human failure. In general, instrumentation failure is less likely to occur when compared to human failure; thus, human failure is the dominant contributor to the failure mechanism in this case. As suggested in *Analysis of Mechanisms for Early Waste Package/Drip Shield Failure* (SNL 2007, p. 6-37), a representative human failure event in this case would be "improperly reading and recording a digital display" with a mean value of 1.25×10^{-3}.

In addition, there are alternatives that can help reduce the probability of using inappropriate material for neutron absorption during the manufacturing and assembly of a TAD canister basket such as analysis of metal coupons.

The use of a proper and specified combination of instrumentation available at the time of manufacturing together with a rigorous quality assurance process, justifies the reduction in material selection error of neutron absorbers by at least three orders of magnitude.

As stated in the response to RAI 4.2.5.1-1-009, prior to acceptance of commercial spent nuclear fuel, the DOE will perform reviews of commercial licensees and certificate holder quality-affecting work-related documentation 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 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. As described in SAR Section 5.10.2.4.2 and Table 5.10-3, these administrative controls will be compiled in the Technical Requirements Manual, 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 and canisters are acceptable from a postclosure criticality perspective. The administrative controls described in SAR Section 5.10.2 and SAR Table 5.10-3 require that similar analyses be completed prior to receiving individual waste forms or canisters/waste package design configurations that are not explicitly analyzed in the license application.

2. COMMITMENTS TO NRC

None.

3. DESCRIPTION OF PROPOSED LA CHANGE

None.

4. REFERENCES

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 7:

Provide information on variability and uncertainty for the non-bounding point-value parameters used in criticality FEPs screening analyses, as required in 10 CFR 63.114(b). Discuss how variability and uncertainty would impact the calculated probability (see ANL–WIS–MD–000027).

1. RESPONSE

In summary, the NRC states in *Yucca Mountain Review Plan, Final Report* (NRC 2003), "…the mean of the distribution range is to be used to screen an event from the performance assessment." Hence, the criticality event class can be properly screened out using a bounding estimate of the mean probability of criticality. Consequently, the DOE does not use distributions for non-bounding point-value parameters or present a distribution for the probability of criticality in the SAR, but in this response the DOE demonstrates how the probability of criticality would decrease if uncertainty in parameters were explicitly represented, rather than accounted for by a bounding representation.

In concept, aleatory uncertainty regarding future events, variability among waste forms, and epistemic uncertainty of parameters impact the calculated probability of criticality. In general, the mean probability of criticality is estimated by dividing the Criticality Event Class (SAR Section 2.2.1.4.1) into several scenarios (reflecting aleatory uncertainty). Variability in package loading and waste form is considered by identifying several loading conditions and waste forms. For each combination of scenario, waste package loading condition, and waste form/package type, a probability of criticality is estimated.

The sum of the probabilities of criticality over all scenarios is compared with the regulatory low-probability criterion to screen a feature, event, or process (FEP) or scenario class where the NRC states in 10 CFR 63.342(a), (SAR Section 2.2.1.2): "DOE's performance assessments… shall not include consideration of very unlikely features, events, or processes, i.e., those that are estimated to have less than one chance in 10,000 of occurrence within 10,000 years of disposal (less than one chance in 100,000,000 per year of occurring)."[1] The NRC elaborates in *Yucca Mountain Review Plan, Final Report* (NRC 2003, Review Method 5, Uncertainty in Event Probability, p. 2.2-14), and states, "Note: Although probability distributions or ranges can include probabilities less than 10^{-8} per year, the mean of the distribution range is to be used to screen an event from the performance assessment."

The mean of the distribution of the probability of criticality could be estimated using mean values for uncertain parameters (because the probability is a linear combination of underlying

[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. 10811. The final rule is effective April 13, 2009 and does not contain any material differences from the proposed rule with respect to this response.

parameters); however, the DOE estimated a bound on the mean by using a combination of mean and bounding values for uncertain parameters. Hence, the DOE screening analysis accounts for variability and uncertainty but does not report a distribution for the probability of criticality.

In the following sections, the DOE elaborates upon the calculation of the bound on mean probability of criticality for the waste form, scenario, and uncertain parameters that most influence the total probability of criticality.

1.1 INFLUENCE OF VARIABILITY AND UNCERTAINTY ON PROBABILITY OF CRITICALITY

1.1.1 Scenario Class Screening

To screen the Criticality Event Class using the low probability criterion, the DOE must show that the mean probability of the Criticality Event Class is less than 10^{-4} over 10,000 years. The DOE defines the Criticality Event Class as the occurrence of one or more critical events in the repository during the post-closure period. To evaluate the probability of this event class, the DOE divided the Criticality Event Class into sixteen scenarios (also called FEP scenarios) distinguished by the aleatory uncertainty in events (categorized as nominal, rockfall, seismic, and igneous (SAR, Section 2.2.1.4.1)) and by three locations for the potential criticality (categorized as inside a package with intact internal conditions, inside a package with degraded internal conditions, in the near-field, and in the far-field). The latter scenario category is suggested by the NRC in Section 2.2.1.2.2.3 of *Yucca Mountain Review Plan, Final Report* (NRC 2003, Acceptance Criterion 1 (2)): "…Criticality events are calculated separately by location."

To facilitate computation of the probability of criticality in the seismic scenarios, the seismic initiating event was further subdivided into three cases distinguished by the effect of the seismic event: vibratory ground motion damage to waste packages, vibratory ground motion disruption of drip shields, or fault displacement (SNL 2008, Section 6.4).

Furthermore, variability in waste package loading was considered by identifying three loading conditions: one correct loading condition (packages with fuel and neutron absorbers loaded as designed) and two misloaded conditions (packages with neutron absorbers misloaded and packages with fuel loaded with insufficient burn-up). Finally, variability from 11 waste forms was evaluated (i.e., nine categories of DOE spent nuclear fuel (SNF) and two categories of commercial SNF) (SNL 2008, Table 4.1-2).

For each combination of scenario, waste package loading condition, and waste form/package type, the analysis reported in *Screening Analysis of Criticality Features, Events, and Processes for the License Application* (SNL 2008) discusses the potential for a critical event to occur. For those combinations with potential for criticality, the analysis computes either an estimate of the mean probability of criticality (using mean values for uncertain parameters) or a bound on the mean probability of criticality (using bounding values for uncertain parameters). This approach ensures that the resulting sum over all scenarios bounds the total mean probability of criticality.

1.1.2 Results of Criticality Screening Analysis

As summarized in SAR Table 2.2-8, the bound on the total mean probability of criticality, considering all disposal waste forms, is 4.4×10^{-5} over 10^4 years. Probability values of criticality for commercial and DOE SNF were estimated only for the in-package degraded configuration, because all other configurations were determined to have an insignificant potential for criticality (SNL 2008, Table 7.1-1). Moreover, for the in-package degraded location, the largest contribution to the total mean probability of criticality (3.7×10^{-5} over 10^4 years) is from seismic ground motion damage to packages with high-level waste (HLW) glass canisters and DOE SNF canisters (codisposal packages). Thus, the effect of variability and uncertainty on the total mean probability of criticality can be assessed from examining this particular scenario.

1.1.3 Probability of Criticality in Degraded Package from Seismic Ground Motion

The probability of criticality inside a degraded package when seismic ground motion is the initiating event breaching the package is the product of three factors (where the second and third factors are combined in a binomial distribution to determine the probability of criticality for one or more waste packages):

1. Probability of one or more waste packages damaged by seismic ground motion
2. Probability of one misloaded codisposal waste package
3. Probability of an in-package criticality given a misloaded waste package with degraded internals and breached outer corrosion barrier.

Concerning the first factor, the probability that seismic ground motion events result in damage to one or more codisposal waste packages within 10,000 years was estimated to be bounded by 0.24 by the method described in *Screening Analysis of Criticality Features, Events, and Processes for the License Application* (SNL 2008, Section 6.4.2.1). This value results from averaging over the aleatory uncertainty in seismic events, which is expressed by the mean seismic hazard curve for bounded ground motion (SAR Figure 2.3.4-18). Given a seismic event, this average probability of damage to waste packages is also uncertain (SAR Table 2.3.4-30). Uncertainty in this probability results from epistemic uncertainty in the residual stress threshold (expressed as a percent of yield strength) for Alloy 22. The criticality screening analysis was performed by assuming the lowest value (90%) for the residual stress threshold. Thus, the value of 0.24 is an upper bound on the probability of damage to DOE codisposal waste packages. As stated in the SAR (p. 2.2-54), this bound is larger than the mean probability by about a factor of three. Thus, if the uncertainty in the residual stress threshold were explicitly included (rather than accounted for by means of a bounding assumption), the estimated probability of damage to waste packages would decrease, which would lead to a decrease in the probability of criticality.

Concerning the second factor, misloading of waste forms in DOE SNF canisters is very improbable because the shape and size of the HLW glass canisters and the various DOE SNF canisters differ significantly and can be readily distinguished by visual inspection. Consequently, misloaded codisposal waste packages could result only by misloading the

absorber plate materials in the package As discussed in *Analysis of Mechanisms for Early Waste Package/Drip Shield Failure* (SNL 2007, Section 6.3.2) (with further justification in the responses to RAIs 3.2.2.1.2.1-4-1 to 3.2.2.1.2.1-4-3), a material selection error was modeled as the product of two factors, each with an error factor of three, with a resulting mean probability of 1.25×10^{-7} (SNL 2008, Section 6.3.2, Table 4.1-1) and 5th and 95th percentiles of 2.6×10^{-8} and 5.9×10^{-7}, respectively.

Concerning the third factor, the probability of in-package criticality given a misloaded waste package for a codisposal package with degraded internals and breached outer corrosion barrier was bounded by one. This bounding assumption is a conservative simplification for the criticality screening analysis. Other analyses have shown that neutron poisoning is only required for the degraded scenario for a limited number of DOE SNF in a fully flooded package (Radulescu et al. 2004; DOE 2004, p. 43). Clearly, the probability of having a fully flooded package is less than one. In addition, significant fuel degradation and/or fuel reconfiguration is required to achieve a critical configuration for codisposal packages for which the probability is also less than one. Therefore, the explicit inclusion of these probabilities would reduce the resulting probability of an in-package criticality given a misloaded waste package to less than one, and most likely, much less than one.

The probability of criticality in one misloaded package (p) is then the product of 1.25×10^{-7} and 1.0. The probability of criticality in one or more misloaded packages is evaluated from the probability of one minus the probability of zero criticality in misloaded packages where the latter probability is evaluated by a binomial distribution with zero critical waste packages, which, in turn, is approximated by a Poisson distribution (SNL 2008, Eq. 6.3-2):

$$1 - P_B(k=0; p, N) \cong 1 - P_P(k=0; pN) = 1 - \frac{(pN)^k e^{-pN}}{k!} = 1 - e^{-pN} \cong pN \qquad \text{(Eq. 1)}$$

where k is the number of critical packages (zero), N is the number of codisposal packages with potential for criticality if misloaded (1,223) (SNL 2008, Section 6.4.2.1), and p is the probability (per package) of criticality. The final approximation in Equation 1 is based on a first-order Taylor series estimation of an exponential and valid for small pN. Because the approximation in Equation 1 is linear in p, evaluation of Equation 1 with the probability of a critical package, $p = 1.25 \times 10^{-7}$, yields the probability of one or more critical packages of $(1.25 \times 10^{-7}) \times 1{,}223 = 1.53 \times 10^{-4}$. At this point, a range of uncertainty is not meaningful in that the result includes the bounding probability of criticality given a misload equal to one.

The bounded mean probability of criticality for the seismic ground motion scenario at the in-package location over all codisposal packages in the repository is then 3.7×10^{-5} over 10^4 years, which is the product of the bounding value 0.24 and bounding value 1.53×10^{-4}.

1.2 CONCLUSIONS

In *Yucca Mountain Review Plan, Final Report* (NRC 2003), the NRC states, "…the mean of the distribution range is to be used to screen an event from the performance assessment." Hence, the

criticality event class can be properly screened out using a bounding estimate of the mean probability of criticality.

In this response, the DOE describes the method for computing the mean probability of criticality as the product of several factors and describes how variability and uncertainty affects the calculated mean probability of criticality. The DOE demonstrates that the probability of criticality would decrease if uncertainty in parameters were explicitly represented, rather than accounted for by a bounding representation.

2. COMMITTMENTS TO NRC

None.

3. DESCRIPTION OF PROPOSED LA CHANGE

None.

4. REFERENCES

DOE (U.S. Department of Energy) 2004. *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.

Radulescu, H.; Moscalu, D.; and Saglam, M. 2004. *DOE SNF Phase I and II Summary Report*. TDR-DSD-MD-000001 REV 00. Las Vegas, Nevada: Bechtel SAIC Company. ACC: DOC.20040303.0005.

NRC 2003. Yucca *Mountain Review Plan, Final Report.* NUREG-1804, Rev. 2, Washington, D.C.: U.S. Nuclear Regulatory Commission, Office of the Nuclear Material Safety and Safeguards.

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; DOC.20071003.0015.

SNL 2008. *Screening Analysis of Criticality Features, Events, and Processes for the 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 8:

Justify that the presented CSNF loading curves for hypothetical canisters would bound the actual canister designs since variations in basket design parameters can have a large impact on the loading curve (see ANL-EBS-NU-000010 REV00)

1. RESPONSE

The commercial spent nuclear fuel (CSNF) loading curves are based on the design basis configuration described in SAR Section 2.2.1.4.1.1.2.1. Section 3.1.5 of *Transportation, Aging and Disposal Canister System Performance Specification* (DOE 2008) provides the postclosure criticality requirements for the transportation, aging, and disposal (TAD) canister design. The key canister design parameter that the design basis configuration takes into account is the borated stainless steel absorber. The postclosure criticality requirement for the borated stainless steel absorber is that the minimum neutron absorber thickness must be based on a corrosion rate of 250 nm/yr for all surfaces of the absorber such that after 10,000 years of corrosion, no less than 6 mm of borated stainless steel will remain separating assemblies. The design basis configuration does not credit separation between the assemblies, nor does it credit any other material.

Since TAD canisters are required to comply with Section 3.1.5 of *Transportation, Aging and Disposal Canister System Performance Specification* (DOE 2008), actual TAD canister designs are expected to be bounded by the loading curves for the design basis configuration.

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. As described in SAR Section 5.10.2.4.2 and Table 5.10-3, these administrative controls will be compiled in the Technical Requirements Manual, 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 and canisters are acceptable from a postclosure criticality perspective. The administrative controls described in SAR Section 5.10.2 and SAR Table 5.10-3 require that similar analyses be completed prior to receiving individual waste forms or canisters/waste package design configurations that are not explicitly analyzed in the license application.

2. COMMITMENTS TO NRC

None.

3. DESCRIPTION OF PROPOSED LA CHANGE

None.

4. REFERENCES

DOE (U.S. Department of Energy) 2008. *Transportation, Aging and Disposal Canister System Performance Specification.* WMO-TADCS-000001, Rev. 1 ICN 1. Washington, D.C.: U.S. Department of Energy, Office of Civilian Radioactive Waste Management. ACC: DOC.20080331.0001.

RAI: Volume 3, Chapter 2.2.1.2.1, Fourth Set, Number 9:

Justify that the parameters of the hypothetical canisters that have been analyzed in loading curve development are bounding for the actual canisters to be used for loading (see ANL-EBS-NU-000010 REV00).

1. RESPONSE

The commercial spent nuclear fuel (SNF) loading curves are based on the design basis configuration described in SAR Section 2.2.1.4.1.1.2.1. Section 3.1.5 of *Transportation, Aging and Disposal Canister System Performance Specification* (DOE 2008) provides the postclosure criticality requirements for the transportation, aging, and disposal (TAD) canister design. The key canister design parameter that the design basis configuration takes into account is the borated stainless steel absorber. The postclosure criticality requirement for the borated stainless steel absorber is that the minimum neutron absorber thickness must be based on a corrosion rate of 250 nm/yr for all surfaces of the absorber such that after 10,000 years of corrosion, no less than 6 mm of borated stainless steel will remain separating assemblies. The design basis configuration does not credit separation between the assemblies, nor does it credit any other material.

Since TAD canisters are required to comply with Section 3.1.5 of *Transportation, Aging and Disposal Canister System Performance Specification* (DOE 2008), actual TAD canister designs are expected to be bounded by the loading curves for the design basis configuration.

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. As described in SAR Section 5.10.2.4.2 and Table 5.10-3, these administrative controls will be compiled in the *Technical Requirements Manual*, 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 and canisters are acceptable from a postclosure criticality perspective. The administrative controls described in SAR Section 5.10.2 and SAR Table 5.10-3 require that similar analyses be completed prior to receiving individual waste forms or canisters/waste package design configurations that are not explicitly analyzed in the license application.

2. COMMITMENTS TO NRC

None.

3. DESCRIPTION OF PROPOSED LA CHANGE

None.

4. REFERENCES

DOE (U.S. Department of Energy) 2008. *Transportation, Aging and Disposal Canister System Performance Specification.* WMO-TADCS-000001, Rev. 1 ICN 1. Washington, D.C.: U.S. Department of Energy, Office of Civilian Radioactive Waste Management. ACC: DOC.20080331.0001.

RAI: Volume 3, Chapter 2.2.1.2.1, Fourth Set, Number 11:

Justify using the 0.001 value assigned to the human error probability CHECK_BM_FLAW top event in (ANL–EBS–MD–000076 Rev 00, Section 6.3.2) from NUREG-1278 Table 20-22 (Item 2) that estimates HEP for error of commission in reading digital display instead of Item 4 from the same table: "Checking that involves active participation, such as special measurements". These measurements are described as checking in nature, involve active measurements, and are described in (ANL-EBS-MD-000076 Rev 00, Section 6.3.2) as follows: "Quick field measurements of material compositions" performed by "the technician assigned to this work".

1. RESPONSE

This RAI requests additional justification for the value assigned to the human error probability for the CHECK_BM_FLAW event used in computing the probabilities of early failure of waste packages (SNL 2007, Section 6.3.2).

Section 1.1 discusses the programmatic ties between the postclosure criticality analysis and preclosure/off-site operations. Section 1.2 provides justification for the values for the human error probability for the CHECK_BM_FLAW event in *Analysis of Mechanisms for Early Waste Package/Drip Shield Failure* (ANL–EBS–MD–000076 Rev 00) (SNL 2007).

1.1 PRECLOSURE/OFF-SITE OPERATION AND POSTCLOSURE CRITICALITY

The information requested by this RAI pertains to obtaining reliability values for geologic repository operational area (GROA) operations during preclosure and/or off-site activities and processes 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 surrogate processes. The surrogate analyses 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. DOE will implement appropriate administrative controls to ensure that the reliability limits used in the preclosure and postclosure safety analyses are met.

As stated in the response to RAI 4.2.5.1-1-009, prior to acceptance of commercial spent nuclear fuel (SNF), the DOE will perform reviews of commercial licensees and certificate holder quality-affecting work-related documentation to ensure that the work has been performed in a manner that complied 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 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.

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 DOE in accordance with the requirements of the license specifications. The Technical Requirements Manual will contain information necessary to support and implement programs listed in the administrative controls section of the license specifications that are unique to the geologic repository, such as the Waste Form and Waste Package Qualification Program and the Waste Package Loading, Handling, and Emplacement 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 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 and SAR Table 5.10-3 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 to the NRC prior to the NRC's issuance of a license to receive and possess SNF and high-level waste (HLW). The final 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 VALUES USED FOR CHECK_BM_FLAW

DOE notes that the reference to "Table 20-22 (Item 2)" in NRC's request is not correct. In *Analysis of Mechanisms for Early Waste Package/Drip Shield Failure* (SNL 2007), the human error probability for the event CHECK_BM_FLAW was modeled by Item 2 of Table 20-10 in NUREG-1278 (Swain and Guttman 1983), not as Item 2 of Table 20-22. NRC suggests that DOE could have selected Item 4 of Table 20-22 in NUREG-1278 to model this event.

In RAI 3.2.2.1.2.2-003, NRC requested DOE to "Justify use of the following data, given in *Analysis of Mechanisms for Early Waste Package/Drip Shield Failure* (SNL 2007a), to calculate the early failure probabilities:

(a) Use of the value from Swain and Guttmann (1983, Item 2, Table 20-10) for the events

(i) CHECK_BM_FLAW in the event tree for evaluating improper base metal selection for the waste package outer corrosion barrier."

DOE's response to RAI 3.2.2.1.2.2-003 addresses this RAI and is repeated here.

Although the CHECK_BM_FLAW top event includes the word "CHECK" in its label, the event is not representing a check in the sense of that relevant to Table 20-22 in *Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications Final Report* (Swain and Guttmann 1983 (i.e., checker detecting errors made by others)). Rather, the event refers to a measurement of the chemical composition of the waste package base metal by the material supplier above and beyond that done in support of preparing a material certification. Such an inspection is consistent with SAR Tables 1.5.2-7 and 1.9-9 (postclosure (design) control parameter 03-19) and in compliance with 10 CFR 63.142. This inspection might be accomplished with an X-Ray spectrometer (SNL 2007, Section 6.3.2) or other instrument. Since this confirmation is performed by an independent organization and procedure, it is appropriately represented by a human error probability of improperly reading and recording a digital display (i.e., a nominal (median) probability of 0.001 and an error factor of 3) (Swain and Guttmann 1983, Table 20-10).

Because the fabrication and handling of the waste package outer corrosion barrier will be accomplished under stringent controls (SAR Table 1.9-9, waste package design control parameters) and in accordance with standard nuclear industry practices, requirements, and procedures (e.g., *ASME Boiler and Pressure Vessel Code* (ASME 2001, Section III, Division 1, Subsection NC) for Class 2 components), it is expected that the material supplier and/or fabricator would check the results of the measurement of the chemical composition of the waste package base metal (accomplished in support of preparing a material certification) and the Yucca Mountain Project would check the material certifications and the fabricator's records upon receipt. Thus, the probability of the base metal being acceptable (BM_FLAW) upon receipt by the Yucca Mountain Project is expected to be higher than is being represented in the event tree. For these reasons, the treatment is appropriate and justified.

2. COMMITMENTS TO NRC

None.

3. DESCRIPTION OF PROPOSED LA CHANGE

None.

4. REFERENCES

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.

Swain, A.D. and Guttmann, H.E. 1983. *Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications Final Report*. NUREG/CR-1278. Washington, D.C.: U.S. Nuclear Regulatory Commission.

RAI: Volume 3, Chapter 2.2.1.2.1, Fourth Set, Number 12:

Justify associating this failure probability solely with a human error when procedure errors, reactor record errors, equipment malfunction, and fuel fabrication/indeterminate errors might also contribute.[1]

1. RESPONSE

This response provides additional justification for basing the probability of a loading curve violation on human error solely in *Commercial Spent Nuclear Fuel Waste Package Misload Analysis* (BSC 2003). As specified in the lead-in statement to this RAI, the context of the response is to justify the probability of a loading curve violation of 1.18×10^{-5} per waste package. This RAI and the response are only relevant to pressurized water reactor (PWR) spent nuclear fuel (SNF) because the selected boiling water reactor (BWR) SNF inventory for the repository is in the acceptable range of the loading curve (SAR Section 2.2.1.4.1.3.2, p 2.2-50), and therefore there is no potential for misloading a BWR transportation, aging, and disposal (TAD) canister.

This RAI response pertains to obtaining reliability values for preclosure geologic repository operational area (GROA) and off-site operations and processes. These processes include SNF loading operations. Because loading operations have yet to be developed, postclosure criticality analyses are based on reliability values calculated for surrogate processes involving generic administrative controls. Surrogate loading processes are described in Section 6.1.3 of *Commercial Spent Nuclear Fuel Waste Package Misload Analysis* (BSC 2003). Analyses based on these surrogate processes establish reasonable reliability limits and provide confidence that, once final loading operations have been established, the postclosure criticality analysis conclusions will remain valid. The DOE will implement appropriate administrative controls with sufficient reliability that the limits used in the postclosure criticality analysis are met.

The input sources for the misload calculations documented in *Commercial Spent Nuclear Fuel Waste Package Misload Analysis* (BSC 2003, Section 4) include human error probabilities and industry data for procedure errors and fuel fabrication/indeterminate errors based on fuel assembly movements and misloads. As described in Section 6.3.1 of *Commercial Spent Nuclear Fuel Waste Package Misload Analysis* (BSC 2003), based on industry data, 327 misloads occurred in approximately 1.2 million fuel assembly movements. Table 1 provides the industry breakdown of the cause of the misloads along with the associated number of misloads.

[1] This RAI is in a group of eight RAIs pertaining to the probability of loading curve violation per canister (i.e., 1.18×10^{-5}) as calculated in BSC 2003.

Table 1. Fuel Assembly Misloads Based on Industry Data

Misload Group	Number of Fuel Assemblies Affected
Misload due to Human Error	225
Misload due to Procedure Errors	11
Misload due to Fabrication/Indeterminate Errors	91

Source: BSC 2001, Table 3.

Two types of analyses are documented in *Commercial Spent Nuclear Fuel Waste Package Misload Analysis* (BSC 2003) as follows:

(1) An analysis based solely on human reliability, which resulted in a misload probability of 1.18×10^{-5} per canister (BSC 2003, Section 6.2.1, Table 13)

(2) A sensitivity analysis based on the industry data listed in Table 1 which resulted in a misload probability of 6.7×10^{-6} per canister (BSC 2003, Section 6.3.1, Table 20). For this sensitivity analysis, a probability distribution for selecting the wrong assembly was obtained using a Bayesian approach based on the data listed in Table 1. This probability distribution replaced the operator selection error in the event tree analysis (BSC 2003, Figure I-1, p. I-7).

The more limiting misload probability of 1.18×10^{-5} per canister was conservatively used as the basis for the postclosure criticality screening analysis.

As stated in the response to RAI 4.2.5.1-1-009, prior to acceptance of commercial spent nuclear fuel, the DOE will perform reviews of commercial licensees and certificate holder quality-affecting work-related documentation 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 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. As described in SAR Section 5.10.2.4.2 and Table 5.10-3, these administrative controls will be compiled in the Technical Requirements Manual, 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 and canisters are acceptable from a postclosure criticality perspective. The administrative controls described in SAR Section 5.10.2 and SAR Table 5.10-3 require that similar analyses be completed prior to receiving individual waste forms or canisters/waste package design configurations that are not explicitly analyzed in the license application.

2. COMMITMENTS TO NRC

None.

3. DESCRIPTION OF PROPOSED LA CHANGE

None.

4. REFERENCES

BSC (Bechtel SAIC Company) 2001. *Waste Package Misload Probability.* CAL-WHS-MD-000001 REV 00. Las Vegas, Nevada: Bechtel SAIC Company. ACC: MOL.20011212.0186.

BSC 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.

RAI: Volume 3, Chapter 2.2.1.2.1, Fourth Set, Number 14:

Justify the assumption made that all shipping records for the fuel assemblies are correct and do not lead to underestimation of misloading events. (Bullet 5.5, page 12 of CAL-WHS-MD-000003).[1]

1. RESPONSE

This response provides justification of the assumption in *Commercial Spent Nuclear Fuel Waste Package Misload Analysis* (BSC 2003) that all shipping records for the fuel assemblies are correct and do not lead to underestimation of misloading events. As specified in the lead-in statement to this RAI, the context of the response is to justify the probability of a loading curve violation of 1.18×10^{-5} per waste package. This RAI and the response are only relevant to pressurized water reactor (PWR) spent nuclear fuel (SNF) because the selected boiling water reactor (BWR) SNF inventory for the repository is in the acceptable range of the loading curve (SAR Section 2.2.1.4.1.3.2, p. 2.2-50), and therefore there is no potential for misloading a BWR transportation, aging, and disposal (TAD) canister.

This RAI response pertains to obtaining reliability values for preclosure geologic repository operational area (GROA) and off-site operations and processes. These processes include SNF loading operations. Because loading operations have yet to be developed, postclosure criticality analyses are based on reliability values calculated for surrogate processes involving generic administrative controls. Surrogate loading processes are described in Section 6.1.3 of *Commercial Spent Nuclear Fuel Waste Package Misload Analysis* (BSC 2003). Analyses based on these surrogate processes establish reasonable reliability limits and provide confidence that, once final loading operations have been established, the postclosure criticality analysis conclusions will remain valid. The DOE will implement appropriate administrative controls with sufficient reliability that the limits used in the postclosure criticality analysis are met.

The TAD canister concept assumes loading approximately 90% of the SNF into TAD canisters at SNF custodian sites. Therefore, loading curve violations for 90% of the TAD canisters are subject to off-site operations (at the SNF custodian sites), and only 10% of the TAD canisters are subject to misloads during GROA operations.

For PWR TAD canisters loaded at the SNF custodian sites, shipping records are not applicable to the misload probability because the TAD canisters are maintained sealed during operations within the GROA, and because the SNF custodian site-based probability of misload per TAD canister remains unchanged.

For TAD canisters loaded within the GROA, reliability of shipping records will be taken into account in the misload analysis and, if necessary, controls will be implemented to demonstrate that the misload probability for all TAD canisters will not exceed 1.18×10^{-5} per TAD canister

[1] This RAI is in a group of eight RAIs pertaining to the probability of loading curve violation per canister (i.e., 1.18×10^{-5}) as calculated in BSC 2003.

once it has been determined which sites and what assemblies will be shipped to the repository in transportation casks rather than in sealed TAD canisters.

As stated in the response to RAI 4.2.5.1-1-009, prior to acceptance of commercial spent nuclear fuel, the DOE will perform reviews of commercial licensees and certificate holder quality-affecting work-related documentation 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 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. As described in SAR Section 5.10.2.4.2 and Table 5.10-3, these administrative controls will be compiled in the Technical Requirements Manual, 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 and canisters are acceptable from a postclosure criticality perspective. The administrative controls described in SAR Section 5.10.2 and SAR Table 5.10-3 require that similar analyses be completed prior to receiving individual waste forms or canisters/waste package design configurations that are not explicitly analyzed in the license application.

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.

RAI: Volume 3, Chapter 2.2.1.2.1, Fourth Set, Number 18:

Justify the assumption (Bullet 5.4) that "only uncanistered fuel is loaded into the waste packages for this calculation, and all canistered fuel that is shipped to the repository will be loaded into a waste package upon receipt."

1. RESPONSE

"Canistered fuel" in the context of this assumption refers to damaged fuel cans, which are readily identifiable compared to intact fuel assemblies and cannot be misloaded. Disposal of damaged fuel cans is not part of the current analysis basis for screening of postclosure criticality. As stated in SAR Section 1.5.1.1.1.2.1.1, notification from the generator is required prior to shipping damaged fuel cans, and plans for handling damaged fuel cans will be developed on an individual basis. The intent is to evaluate the damaged fuel can and its contents prior to allowing shipment to the repository.

SAR Section 2.2.1.4.1 presents the methodology and analyses required to confirm that waste forms and canisters are acceptable from a postclosure criticality perspective. The administrative controls described in SAR Section 5.10.2 and SAR Table 5.10-3 require that similar analyses be completed prior to receiving individual waste forms or canisters/waste package design configurations that are not explicitly analyzed in the license application.

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.

RAI: Volume 3, Chapter 2.2.1.2.1, Fourth Set, Number 20:

Justify not considering other events in the probability calculations in addition to misloading and manufacturing errors, such as reactor records errors: basket degradation, preferential absorber degradation due to handling damage (developed holes in plates/tubes), fuel/absorber reconfiguration due to the effects of oxide wedging, and accumulation of hydrated corrosion products, separation of absorber plates/tubes that are mechanically assembled. The calculation only considers assembly and neutron absorber misloads as contributors to the potential of criticality (see ANL–DS0–NU–000001).

1. RESPONSE

As stated in *Screening Analysis of Criticality Features, Events, and Processes for License Application* (SNL 2008a, Section 1), an evaluation of the criticality features, events, and process (FEP) scenarios identifies misloading and manufacturing errors as two common events that were present in each in-package event sequence with a potential for criticality. In addition to the two common events, other events have been considered and implemented in the design basis configurations. Therefore, these events have been assigned a probability of one of occurring.

The following sections clarify how the specific events listed in the RAI have been considered and implemented in the design basis configurations.

1.1 REACTOR RECORD ERRORS

Reactor record errors are accounted for with a 5%-burnup uncertainty penalty applied to the loading curve as explained in SAR Section 2.2.1.4.1.1.4. The DOE expects commercial licensees/certificate holders will maintain and implement a quality assurance program consistent with applicable NRC regulations, 10 CFR 50, 71, or 72. These records are relied upon for other NRC-regulated operations such as reactor core reload licensing evaluations, spent fuel pool evaluations, on-site dry cask storage evaluations, and as the basis for the nuclear material accountability reports (Forms NRC 741/742) filed with the NRC as required by 10 CFR 74. Administrative controls (SAR Section 5.10.3) will be implemented to ensure the traceability of the reactor records. For transportation, aging, and disposal (TAD) canisters loaded at the spent nuclear fuel (SNF) custodian sites, reactor engineering data acquired under custodian 10 CFR 50 QA programs are expected to be available and able to demonstrate that the records are reliable to ≤ 5% of the reported burnup values. As described in the response to RAI 4.2.5.1-1-009, prior to acceptance of commercial SNF, the DOE will perform reviews of the commercial licensee and certificate holders quality-affecting, work-related documentation to ensure work has been performed in a manner that complies with applicable quality assurance requirements and, where applicable, the DOE standard contract.

1.2 BASKET DEGRADATION

Basket degradation is accounted for in the development of the design basis configurations as explained in SAR Section 2.2.1.4.1.1.2.2. For commercial SNF packages, a simulated

10,000 years of corrosion has been applied to the neutron absorber plates within the package resulting in a reduced material thickness applied to the design basis configuration. For the DOE SNF canister basket designs, the absorber is only necessary when the waste form and internal components degrade into certain limiting configurations. Therefore, basket degradation is accounted for. For certain degraded configurations where additional absorber material is needed to suppress the reactivity of the system for certain configurations, the material is interspersed within the canister to intermix with the waste form as it degrades over time.

1.3 PREFERENTIAL ABSORBER DEGRADATION

Preferential absorber degradation due to handling damage (developed holes in plates/tubes) is accounted for within the absorber misload event. The neutron absorber misload event represents a basket fabrication error resulting in improper performance of the neutron absorber plates (e.g., incorrect material installed during fabrication or absorber content of plates outside specified range). As identified in SAR Table 5.10-3, row: *Waste Package Loading, Handling, and Emplacement Program*, administrative controls will be established to control the loading, handling, and emplacement of waste packages. The waste package loading, handling, and emplacement program will include provisions to ensure that the waste packages are loaded, handled, and emplaced in accordance with assumptions in the total system performance assessment (TSPA).

1.4 OXIDE WEDGING EFFECTS

Fuel/absorber reconfiguration due to the effects of oxide wedging is accounted for with the design basis configuration. The design basis configuration is characterized by certain nonphysical representations that increase system reactivity to bound potential configurations that may occur over time. Effects of oxidation and oxide wedging, as well as the potential relevant configurations that can form inside the canister have been considered in the development of the design basis configuration. These effects are not modeled in the TAD canister design basis configuration because they result in a less reactive configuration as the oxidation materials displace moderator and result in having assemblies in a more spread out geometry. Additionally, collapsing the lattice array of the assembly also reduces the H/X ratio, thereby reducing system reactivity. For the DOE SNF, material reconfiguration is necessary for criticality to be possible. Therefore, detailed material degradation processes are modeled that are waste form and basket specific, and sensitivity analyses are evaluated for developing the most reactive credible combination of materials and geometric orientation for use as the design basis configuration. Oxidation alone is not considered as hydration is a necessary precursor for criticality for the commercial SNF and DOE SNF proposed canister designs.

As noted in the responses for RAIs 3.2.2.1.2.1-4-008 and 3.2.2.1.2.1-4-009, as actual TAD canister designs are developed the processes of corrosion/degradation will be evaluated in accordance with SAR Table 5.10-3, row: *Waste Form and Waste Package Qualification Program*, to ensure the design basis configuration is bounding prior to waste receipt. This administrative control will require that similar analyses following the methodology identified in SAR Section 2.2.1.4.1 be completed prior to receiving individual waste forms or waste package designs that are not explicitly analyzed in the license application.

1.5 HYDRATED CORROSION PRODUCT ACCUMULATION

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, result in an increase in assembly spacing 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 TAD canister specifications for the commercial SNF waste form (DOE 2008). In the design basis configuration model, the regions that would form into a hydrated material are represented as full density water. Sensitivity evaluations have been performed in (SNL 2008b, Section 6.2.5) evaluating the effects of waste form hydration and demonstrating that the design basis configuration is bounding. For the DOE SNF waste forms, the various intact DOE SNF in a flooded canister are critically safe whether the poison is present in the absorber plates or not. The poisoned plates are used as a mechanism to install a measurable quantity of gadolinium needed to reduce the system reactivity for degraded conditions in post-closure. Therefore, these degraded configurations are used to set the criticality control (neutron absorber) requirements. The effects of hydration are a necessary precursor for criticality considerations for the commercial SNF and DOE SNF proposed canister designs. Thus, the effects of hydration and corrosion products have been considered and accounted for when they increase system reactivity.

As noted in the responses for RAIs 3.2.2.1.2.1-4-008 and 3.2.2.1.2.1-4-009, as actual TAD canister designs are developed the processes of corrosion/degradation will be evaluated in accordance with SAR Table 5.10-3, row: *Waste Form and Waste Package Qualification Program*, to ensure the design basis configuration is bounding prior to waste receipt. This administrative control will require that similar analyses following the methodology identified in SAR Section 2.2.1.4.1 be completed prior to receiving individual waste forms or waste package designs that are not explicitly analyzed in the license application.

1.6 SEPARATION OF ABSORBER PLATES/TUBES

Separation of absorber plates/tubes that are mechanically assembled is considered in the design basis configuration. However, the basket configuration and assembly mechanisms for incorporating fixed neutron absorber plates are design dependent. Based on the conceptual TAD canister design evaluated, whether the tubes separate or not, they are expected to remain in their relative positions, as the spacing within the basket will physically limit component movement. The current evaluation is based on the TAD canister performance specifications, which indicate that neutron absorber plates must cover all four longitudinal sides of each fuel assembly (DOE 2008, p. 16). Therefore, the accepted TAD canister designs will ensure that this requirement is met.

As noted in the responses for RAIs 3.2.2.1.2.1-4-008 and 3.2.2.1.2.1-4-009, as actual TAD canister designs are developed the processes of corrosion/degradation and potential for plate separation/relocation will be evaluated in accordance with SAR Table 5.10-3, row: *Waste Form and Waste Package Qualification Program*, to ensure the design basis configuration is bounding

prior to waste receipt. As for the DOE SNF canisters, the use of the nickel-gadolinium alloy is not a preventive approach to criticality for which 'structure' is credited. The various intact DOE SNF waste forms in a flooded canister are critically safe regardless of the presence of poison in the absorber plates (DOE 2004, p. 43). The poisoned plates are used as a mechanism to install a measurable quantity of gadolinium needed to reduce the system reactivity for degraded conditions in postclosure. The degraded condition of the basket plates and potential for redistribution of the gadolinium throughout the degraded SNF canister contents is important to criticality safety for the time period when waste package breaching allows for moderation in and around the DOE SNF canisters (DOE 2004, pp. 26 and 27).

2. COMMITMENTS TO NRC

None.

3. DESCRIPTION OF PROPOSED LA CHANGE

None.

4. REFERENCES

DOE (U.S. Department of Energy) 2004. *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.

DOE 2008. *Transportation, Aging and Disposal Canister System Performance Specification.* WMO-TADCS-000001, Rev. 1 ICN 1. Washington, D.C.: U.S. Department of Energy, Office of Civilian Radioactive Waste Management. ACC: DOC.20080331.0001.

SNL 2008a. *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; DOC.20090302.0002.

SNL 2008b. *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 22:

Provide the calculation of the probability of criticality from possible misloads for realistic worst-case loading operations where risk is not diluted by use of entire fuel inventory (i.e., only assemblies from a particular site are available for selection at that site and overall probability of criticality is determined by the site with the most reactive inventory); or justify that the use of uniform random assembly sampling from the total spent fuel inventory is bounding with respect to criticality for realistic loading operations (0.014, as per (ANL-EBS-NU-000010 REV 00 section 6.4).

1. RESPONSE

This response provides additional justification that use of uniform random assembly sampling from the total spent fuel inventory for calculating the conditional probability of criticality (0.014) given an assembly misload, as opposed to site-specific calculations, does not have a significant impact on screening postclosure criticality from the postclosure performance assessment.

This RAI response pertains to obtaining reliability values for preclosure geologic repository operational area (GROA) and off-site operations and processes. These processes include spent nuclear fuel (SNF) loading operations. Because loading operations have yet to be developed, postclosure criticality analyses are based on reliability values calculated for surrogate processes involving generic administrative controls. Surrogate loading processes are described in Section 6.1.3 of *Commercial Spent Nuclear Fuel Waste Package Misload Analysis* (BSC 2003). Analyses based on these surrogate processes 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 with sufficient reliability that the limits used in the postclosure criticality analysis are met.

This RAI and the response are only relevant to pressurized water reactor (PWR) spent nuclear fuel (SNF) because the selected boiling water reactor (BWR) SNF inventory for the repository is in the acceptable range of the loading curve (SAR Section 2.2.1.4.1.3.2, SAR p. 2.2-50), and, therefore, there is no potential for misloading a BWR transportation, aging, and disposal (TAD) canister.

The probability of criticality due to an assembly misload in a breached PWR waste package is the product of the probability of a misload (1.18×10^{-5} calculated in BSC 2003) and the conditional probability of criticality given a misload (0.014 calculated in SNL 2008a). The conditional probability of criticality in a breached PWR waste package due to a misload ($1.18 \times 10^{-5} \times 0.014 = 1.65 \times 10^{-7}$) is reasonable, and can be met once loading operations have been established.

As described in Section 6.4 of *CSNF Loading Curve Sensitivity Analysis* (SNL 2008a), the conditional probability of criticality given a misload (0.014) is based on the misloaded assembly being inserted in the central location of the PWR TAD canister (i.e., the highest reactivity

contributing location). Taking into account all locations in a PWR TAD canister where an assembly can be misloaded, the probability of criticality decreases to 0.002 (SNL 2008a, Table 6-42). As an additional conservatism, the maximum burnup credit for the surrounding assemblies that would compensate for the reactivity increase due to the misloaded assembly was capped at 50 GWd/MTU, even though over 20% of the projected waste stream is expected to have burnup values greater than 50 GWd/MTU (CRWMS 2000, Attachment III, Disc III, Spreadsheet: *Case_A63_Characteristics.xls*.) Based on these conservatisms, once site-specific inventories have been finalized, and site-specific analyses have been performed, it is reasonable to expect that the probability of criticality in a breached PWR TAD canister due to a misload (1.65×10^{-7}) will not be exceeded.

In addition, even if the probability of criticality due to an assembly misload in a breached PWR waste package is increased from 0.014 to an unrealistic value of 1, the overall probability of postclosure criticality would increase only from 4.4×10^{-5} (SAR Table 2.2-8) to 5.76×10^{-5}, and therefore, postclosure criticality would still remain screened out from the postclosure performance assessment. The probability value of 5.76×10^{-5} can be attained by replacing 0.014 with 1.0 for the probability of criticality due to an assembly misload in a breached PWR waste package in Sections 6.3 and 6.4 in *Screening Analysis of Criticality Features, Events, and Processes for License Application* (SNL 2008b).

As stated in the response to RAI 4.2.5.1-1-009, prior to acceptance of commercial SNF, the DOE will perform reviews of commercial licensees and certificate holder quality-affecting work-related documentation 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 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 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. As described in SAR Section 5.10.2.4.2 and Table 5.10-3, these administrative controls will be compiled in the technical requirements manual, 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 and canisters are acceptable from a postclosure criticality perspective. The administrative controls described in SAR Section 5.10.2 and SAR Table 5.10-3 require that similar analyses be completed prior to receiving individual waste forms or canisters/waste package design configurations that are not explicitly analyzed in the license application.

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.

CRWMS M&O 2000. *Waste Packages and Source Terms for the Commercial 1999 Design Basis Waste Streams*. CAL-MGR-MD-000001 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000214.0479.

DOE 2008. *Transportation, Aging and Disposal Canister System Performance Specification*. WMO-TADCS-000001, Rev. 1 ICN 01. Washington, D.C.: U.S. Department of Energy, Office of Civilian Radioactive Waste Management.

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 28:

Justify the use of nitrate solution benchmarks, given that the most likely form of material will be metal-water or oxide-water systems. It is not clear why these are applicable to disposal conditions.

1. RESPONSE

Nitrate solution benchmarks are not included as a component of the validation methodology documented by *Range of Applicability and Bias Determination for Postclosure Criticality of Commercial Spent Nuclear Fuel* (Radulescu et al. 2007) because preliminary cross-section sensitivity and uncertainty analyses identified those experiments as not having sufficient neutronic similarity to the design-basis waste package configurations for commercial spent nuclear fuel.

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 35:

For DOE SNF, estimate the amount of conservatism resulting from assuming fresh fuel, instead of spent fuel, isotopic abundances.

1. RESPONSE

This question requests an estimate of the amount of conservatism in k_{eff} that is present for the DOE spent nuclear fuel (SNF).

As indicated in SAR Section 1.5.1.3.1.1, there are several hundred distinct types of DOE SNF spanning a wide range of characteristics. Fissile loads can vary from gram to kilogram quantities in a single fuel handling unit. *Packaging Strategies for Criticality Safety for "Other" DOE Fuels in a Repository* (DOE 2004) shows the quantity of the fissile loadings by fuel type. Fissile isotope species include ^{233}U, ^{235}U, and ^{239}Pu with enrichments ranging from depleted uranium up to over 93%-enriched ^{235}U. The DOE SNF has been used in production reactors, fuel performance testing, supporting weapons program, research and training, and for other isotope production programs. The burnup of DOE SNF ranges from slightly irradiated to over 500,000 MWd/MTU (SAR Section 1.5.1.3.1.1, p. 1.5.1-38). The use of beginning of life (BOL) fissile values is more conservative as acknowledged in the RAI, and supports a reasonable approach in addressing those fuels in criticality models. The vast variety of DOE SNF has accumulated since the inception of nuclear reactors. Previously, fuel reprocessing obviated the need to collect and save burnup documentation for those fuels. Yet past handling and current storage practices continue to apply BOL fissile values for management of those fuels. Therefore, an estimate of the amount of conservatism resulting from assuming fresh fuel versus burned fuel is neither readily obtainable nor necessary as the DOE is not requesting burnup credit for DOE SNF waste forms.

In addition to the conservatism provided by using fresh fuel isotopics, the DOE SNF criticality evaluations include the margins estimated below.

For the DOE SNF in a waste package, comprehensive evaluations have been performed to represent the various states of degradation from the intact, as-assembled configuration to fully degraded configurations that may occur over time (Radulescu et al. 2004, Section 6.5). These have been performed using either a fresh fuel assumption or the most reactive fissile concentration (for fuels that contained fertile species that convert to fissile isotopes during irradiation) (Radulescu et al. 2004, Section 5.4.1). Criticality control limits were designed for each representative criticality group based on maintaining subcriticality for the most reactive, credible degraded configurations. Each of the DOE SNF fuel categories used a critical limit of 0.93 for the design basis with the exception of U/Th oxide fuel (represented by Shippingport LWBR), using a value of 0.92 (Radulescu et al. 2004, p. 10-70), and low-enriched uranium oxide fuel (represented by TMI-2), using a value of 0.97 (BSC 2004a, p. 9). The canisters are designed (i.e., allowable fissile load and poisoning requirements) to ensure that the most reactive, credible configurations are below these limits.

Code biases and uncertainties were established using applicable benchmarks to the various configurations. Also, to account for and bound additional uncertainties associated with any of the credible configurations, additional design margin was implemented to establish DOE SNF canister loading limits. Table 1 lists the reference locations for the stated design limits along with the lower-bound tolerance limits accounting for bias and bias uncertainty, and the estimated margins in k_{eff}.

Evaluations for the TMI-2 and ATR representative waste forms are not as complete as for the other DOE SNF fuel groups. The TMI-2 waste form is low-enriched uranium and originated as a commercial fuel assembly prior to the Three Mile Island Unit 2 partial core meltdown. The core debris from TMI-2 was packaged into three different types of canisters and issued certificates of completion (Certificate No. 9200, Docket No. 71-9200) that met transportation requirements prior to shipping for dry storage at Idaho National Laboratory.

Table 1. Source, Bias, and Uncredited Margin Values

Representative Fuel	Subcritical Limit and Reference	Reference Location	Lower- Bound Tolerance Limits [1]	Estimated Margin (Δk_{eff})
FFTF	Radulescu et al. 2004 (0.93)	Section 10.1, p. 10-3	0.9786	0.0486
TRIGA	Radulescu et al. 2004 (0.93)	Section 10.2, p. 10-18	0.9796	0.0496
Shippingport PWR	Radulescu et al. 2004 (0.93)	Section 10.3, p. 10-36	< 0.969 (0 < AENCF < 0.0278) < −0.2336 × AENCF + 0.9755 (0.0278 < AENCF < 0.0922 MeV)	0.024 to 0.039
Enrico Fermi	Radulescu et al. 2004 (0.93)	Section 10.4, p. 10-49	0.9659 to 0.9872	0.0359 to 0.0572
Shippingport LWBR	Radulescu et al. 2004 (0.92)	Section 10.5, p. 10-70	0.9748	0.0548
N Reactor	Radulescu et al. 2004 (0.93)	Section 10.6, p. 10-89	< 0.0765 × AENCF +0.9434 (0 < AENCF < 0.175 MeV) < 0.9568 (AENCF > 0.175)	0.013 to 0.0268
Fort St. Vrain	Radulescu et al. 2004 (0.93)	Section 10.8, p. 10-120	0.9608 (0 < AENCF < 0.4625 MeV) −0.0183 × AENCF + 0.9687 (0.4625 < AENCF < 0.8015 MeV)	0.024 to 0.0308
ATR	BSC 2004b (0.93)	p. 32	0.9659 (applicable to U-Al fuels) (see NOTE)	0.0359 (see NOTE)
TMI-2	BSC 2004a (0.97)	Section 3.6, p. 9	(see NOTE)	(see NOTE)

[1]BSC 2004c, Table 5.

NOTE: The lower-bound tolerance limits and estimated margins for ATR and TMI-2 fuels are explained in the text following this table.

As for the TMI-2 representative fuel shown in Table 1, Babcock & Wilcox performed for the Three Mile Island operator (GPU Nuclear Corporation) a 'comparative analysis' for margin around 1985 when the TMI-2 debris canisters were being loaded with TMI-2 reactor core debris. The analysis considered a 'single, standard configuration' modeling of the knockout canister in KENO, and calculated $k_{eff} + 2\sigma = 0.873$ with B_4C in the canister, and $k_{eff} + 2\sigma = 0.937$ with water replacing B_4C. The model used an average core + 2σ enrichment value, KENO IV with 123

energy group cross sections, and no burnup credit (Babcock & Wilcox 1985, Section 2.4, p. 2-2; and Section 2.5, p. 2-2), and used $k_{eff} < 0.95$ for the criticality safety criterion (Babcock & Wilcox 1985, Section F, p. 36). Baldwin et al. (1979) established the KENO bias with critical experiment measurements made at the Babcock & Wilcox Lynchburg Research Center. *TMI-2 Defueling Canisters Final Design Technical Report* (Babcock & Wilcox 1985, Attachment 2, Section 2.9.2, p. 2-40) concluded, "the comparison between KENO and measured results shows that KENO underpredicted the measured value by less than 0.02 Δk."

The TMI-2 canisters, representing various states of degradation and internal configurations, have already been shipped across the United States using this k_{eff} criterion as applied to a multicanister array in a transport cask. Only single canisters will be loaded per DOE SNF waste package; therefore, sufficient margin has already been incorporated for the loading of the TMI-2 SNF canisters (i.e., 0.05 for transportation).

ATR fuel is currently the representative waste form for the DOE SNF aluminum fuels group. Prior to ATR being designated as the representative waste form, benchmark applicability and bias and uncertainty evaluations were performed for the aluminum fuels group represented by melt and dilute ingots, as documented in *Criticality Model* (BSC 2004c, Attachment VIII), and resulted in a lower-bound tolerance value of 0.9659 (BSC 2004c, Table 5), as shown in Table 1. Although some of these values are not directly applicable to the ATR fuel design, they have characteristics representing the DOE SNF aluminum fuels group. Other high-enriched uranium benchmarks are available in *International Handbook of Evaluated Criticality Safety Benchmark Experiments* (NEA 2008) including one such experiment using ATR fuel (HEU-MET-THERM-022). One would expect an evaluation of other high-enriched uranium HEU critical experiments (that would be used in the bias determination) to result in a lower-bound tolerance limit equal to or greater than that calculated for the melt and dilute ingots.

Applicable benchmark experiments would consist of HEU plate-type fuels as documented in *International Handbook of Evaluated Criticality Safety Benchmark Experiments* (NEA 2008). Table 2 lists three separate example sets consisting of 35 total experiments.

Table 2. Example Benchmarks

Benchmark Experiment	Identifier	Estimated Bias	Reference Location
SPERT-D Aluminum-Clad Plate-Type Fuel in Water, Dilute Uranyl Nitrate, or Borated Uranyl Nitrate	HEU-MET-THERM-006	~1%	p. 38
SPERT III Stainless-Steel-Clad Plate-Type Fuel in Water	HEU-COMP-THERM-022	<1%	p. 32
Advanced Test Reactor: Serpentine Arrangement of Highly Enriched Water-Moderated Uranium-Aluminide Fuel Plates Reflected by Beryllium	HEU-MET-THERM-022	< 0.002 Δk_{eff}	p. 100

Source: NEA 2008.

Based on the example bias results shown in Table 2 compared with the lower-bound tolerance limit in Table 1, a design limit of 0.93 applicable to the aluminum fuels group is conservative with an estimated amount of conservatism on the order of 0.0359 (0.9659-0.93). The establishment of the actual bias and uncertainty will be performed for ATR canisters and confirmed to be bounding prior to waste receipt in accordance with provisions stated in SAR Table 5.10-3 (row: Waste Form and Waste Package Qualification Program). This administrative control will require that similar analyses be completed prior to receiving individual waste forms or waste package designs that are not explicitly analyzed in the license application.

2. COMMITMENTS TO NRC

None.

3. DESCRIPTION OF PROPOSED LA CHANGE

4. REFERENCES

Babcock & Wilcox 1985. *TMI-2 Defueling Canisters Final Design Technical Report*. 77-1153937-04. Lynchburg, Virginia: Babcock & Wilcox.

Baldwin, M.N.; Hoovler, G.S.; Eng, R.L.; and Welfare, F.G. 1979. *Critical Experiments Supporting Close Proximity Water Storage of Power Reactor Fuel.* BAW-1484-7. Lynchburg, Virginia: Babcock & Wilcox.

BSC 2004a. *Intact and Degraded Mode Criticality Calculations for the Codisposal of TMI-2 Spent Nuclear Fuel in a Waste Package*. CAL-DSD-NU-000004 REV 00A. Las Vegas, Nevada: Bechtel SAIC Company. ACC: DOC.20040329.0002.[a]

BSC 2004b. *Intact and Degraded Mode Criticality Calculations for the Codisposal of ATR Spent Nuclear Fuel in a Waste Package*. CAL-DSD-NU-000007 REV 00A. Las Vegas, Nevada: Bechtel SAIC Company. ACC: DOC.20041018.0001.

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

DOE (U.S. Department of Energy) 2004. *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.

NEA (Nuclear Energy Agency) 2008. *International Handbook of Evaluated Criticality Safety Benchmark Experiments.* September 2008 Edition. NEA/NSC/DOC(95)03. Paris, France: Nuclear Energy Agency, Organisation for Economic Co-Operation and Development.

Radulescu, H.; Moscalu, D.; and Saglam, M. 2004. DOE SNF Phase I and II Summary Report. TDR-DSD-MD-000001 REV 00. Las Vegas, Nevada: Bechtel SAIC Company. ACC: DOC.20040303.0005.

NOTE: [a] Provided as an enclosure to letter from Williams to Sulima dated 02/17/2009. "Yucca Mountain – Request for Additional Information Re: License Application (Safety Analysis Report Section 2.1), Safety Evaluation Report Volume 3 – Postclosure Chapters 2.2.1.1 and 2.2.1.3.7 – Submittal of Department of Energy Reference Citations."

RAI: Volume 3, Chapter 2.2.1.2.1, Fourth Set, Number 38:

Explain and justify DOE's assertion in CRWMS M&O (2003) that "there is no risk associated with a subcritical event." This is the only documented justification for the zero margin of subcriticality for postclosure criticality. In particular, explain the following:

a. If the "no risk" refers to the probability of criticality, explain how this is the case, given that criticality calculations are only performed for configurations that cannot be screened out based on probability.

b. If "no risk" refers to the consequences of criticality, explain why this is the case, given that subcriticality is used (as one of several screening criteria) to decide which events will be included in the consequence assessment. This could lead to a circular argument.

c. Justify why a zero margin provides adequate assurance that systems evaluated to be subcritical actually are subcritical. If there is a regulatory reason for criticality calculations to be performed, there is presumably a need to ensure subcriticality with a high degree of confidence.

1. RESPONSE

1.1 EXPLANATION AND JUSTIFICATION OF STATEMENT

The statement, "…there is no risk associated with a subcritical event," was made in the context of describing the equation for the determination of the critical limit and refers to the fact that the consequences of subcritical events are not included in the total system performance assessment (TSPA), as subcritical systems have no impact on repository performance. Specifically, Section 3.5.3.2.5 of *Disposal Criticality Analysis Methodology Topical Report* (YMP 2003, Equation 3-1) describes the critical limit as represented by (based on Section 5 of ANSI/ANS-8.17-1984):

$$CL(x) = f(x) - \Delta k_{EROA} - \Delta k_{ISO} - \Delta k_m \qquad \text{(Eq. 1)}$$

where

CL = critical limit

x = a neutronic parameter used for trending

f(x) = the lower-bound tolerance limit function accounting for biases and uncertainties that cause the calculation results to deviate from the true value of k_{eff} for a critical experiment, as reflected over an appropriate set of critical experiments

Δk_{EROA} = penalty for extending the range of applicability

Δk_{ISO} = penalty for isotopic composition bias and uncertainty

Δk_m = an arbitrary margin ensuring subcriticality for preclosure and turning the critical limit function into an upper subcritical limit function (it is not applicable for use in postclosure analyses because there is no risk associated with a subcritical event).

Equation 1 (YMP 2003, Equation 3-1) is the same as Equation 2.2-1 in SAR Section 2.2.1.4.1.1.2.4.2 As noted above and in SAR Section 2.2.1.4.1.1.2.4.2, Δk_m is not applicable for use in postclosure analyses. The critical limit is the value of k_{eff} at which the configuration class is considered potentially critical as characterized by statistical tolerance limits obtained by analysis of experimental systems with a range of physical and neutronic parameters that are representative of the configuration class parameters expected in the repository. Stated differently, the critical limit is the value of k_{eff} that defines a criticality event for a configuration class. Establishing the critical limit is part of the methodology to determine the probability of a criticality event in the postclosure time period. The probability of criticality is evaluated for those configuration classes for which k_{eff} exceeds the critical limit for the waste form (SAR Figure 2.2-4, Item 2). Subcriticality is a nominal condition and is not evaluated for probability of occurrence.

1.2 EXPLANATION OF RAI PART "A"

If the "no risk" refers to the probability of criticality, explain how this is the case, given that criticality calculations are only performed for configurations that cannot be screened out based on probability.

As described in the introductory text of SAR Section 2.4, "risk" for the postclosure repository system is understood in terms of answering the risk triplet questions: what can happen; how likely is it to happen; and what are the consequences. The proposed 10 CFR 63.2[1] definition of performance assessment directly incorporates the questions of the risk triplet. The reference to, "Identifies features, events, processes" corresponds to the first of the risk triplet questions and "Identifies… their probabilities of occurring" corresponds to the second of the risk triplet questions. Similarly, the 10 CFR 63.2 reference to, "Examines the effects of those features, events, processes" and "Estimates the dose incurred by the reasonably maximally exposed individual" correspond to the third risk triplet question. Given this context, "no risk" does not refer to the probability of criticality, it refers to the condition of a system being subcritical (i.e., k_{eff} <1.0), and how such systems affect the total system performance assessment and dose incurred by the reasonably maximally exposed individual

[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.

(RMEI). There are no additional features, events, or processes associated with a system being subcritical (the packages are loaded, transported, and stored this way) that must be included in the total system performance assessment. Therefore, a system that is subcritical does not present any additional risk that must be accounted for in the postclosure performance assessment.

Criticality calculations are performed to address a variety of needs and are not only performed for configurations that cannot be screened out based on probability. As indicated in SAR Section 2.2.1.4.1.1.2, nuclear criticality validation calculations are performed to determine the critical limit for each scenario class, configuration class, and waste form. k_{eff} values are calculated for relevant configurations and are compared against the critical limit. These types of criticality calculations are performed to support the determination of the probability of occurrence of criticality, not the consequence of criticality. Criticality *consequence* calculations would only be performed if criticality can not be screened out on the probability criterion as proposed in 10 CFR 63.342(a). If the total probability of criticality was equal to or greater than the criterion for event inclusion in the performance assessments, then intermediate criticality consequence analyses (e.g., evaluations to determine the potential for increases in radionuclide inventory, heat generation and subsequent waste heat output affecting degradation of the engineered barrier system or enhanced radionuclide release rates) would then be conducted to determine if criticality needs to be included in the TSPA or can be excluded because of its low consequence to repository performance. The omission of criticality from the TSPA would not significantly change the magnitude and time of radiological releases exposures to the RMEI, or radionuclide releases to the accessible environment pursuant to proposed 10 CFR 63.114(a)(5) and 10 CFR 63.114(a)(6).

1.3 EXPLANATION OF RAI PART "B"

> *If "no risk" refers to the consequences of criticality, explain why this is the case, given that subcriticality is used (as one of several screening criteria) to decide which events will be included in the consequence assessment. This could lead to a circular argument.*

The intermediate consequences of a criticality event with regards to postclosure repository performance that must be considered if criticality is not screened out based on probability are an increase in radionuclide inventory, heat generation, and subsequent waste heat output affecting degradation of the engineered barrier system or enhanced radionuclide release rates. Because subcritical systems cannot sustain a critical reaction, they do not result in either an additional increase in radionuclide inventory or increase the heat in the system above that already accounted for in the TSPA. Thus, subcritical systems have no additional risk to repository performance because there is no change to the repository characteristics in the TSPA.

The consequences of criticality have not been included in the postclosure dose calculations because the probability of a criticality event falls below the regulatory threshold.

1.4 EXPLANATION OF RAI PART "C"

> *Justify why a zero margin provides adequate assurance that systems evaluated to be subcritical actually are subcritical. If there is a regulatory reason for*

> *criticality calculations to be performed, there is presumably a need to ensure subcriticality with a high degree of confidence.*

As described in SAR Section 2.2.1.4.1.1.2.2, to ensure that the calculated k_{eff} value is always greater than the actual k_{eff} value, conservative assumptions and modeling representations are used to define the design basis configurations in performing the criticality calculations for each configuration. The design basis configurations include as-emplaced and degraded configurations. Collectively, these assumptions and representations result in overestimation of the k_{eff} value and provide margin in the analysis predictions and loading curves. Therefore, it is incorrect to state that there is "zero margin"; there is margin in the evaluations of k_{eff} and that margin is sufficient to ensure that there is a high degree of confidence that configurations determined to be subcritical are so. The criticality calculations described in SAR Section 2.2.1.4.1 are performed to address the regulatory requirements of 10 CFR 63.21(c)(9) and 63.102(j), and proposed 10 CFR 63.114 and 63.342. Criticality calculations were performed to support the determination of the probability of occurrence of criticality for comparison with proposed 10 CFR 63.342(a).

As noted in the preamble to 10 CFR 63 (see Federal Register Notice 66 FR 55737), the NRC stated the risk-informed performance-based approach used to develop the rule "eliminates arbitrary or prescriptive siting and design criteria, as well as detailed requirements such as quantitative subsystem performance objectives." 10 CFR 63 does not require the prevention of criticality nor does it establish or require an arbitrary criticality administrative margin (minimum margin of subcriticality) for the postclosure period.

2. COMMITMENTS TO NRC

None.

3. DESCRIPTION OF PROPOSED LA CHANGE

None.

4. REFERENCES

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 41:

> Estimate the effect on the criticality dose if those waste configurations screened out on the basis of subcriticality had not been screened out. That is, determine how much of a difference it would have made to the end result if a subcritical margin of 0.02 were used instead of 0? If a subcritical margin of 0.05 were used?

1. RESPONSE

Consistent with long-term regulatory limits placed on total dose from all significant features, events, and processes, rather than on doses from individual events, the DOE does not estimate dose from criticality events. Criticality is screened from the total system performance assessment in accordance with the probability criterion at proposed 10 CFR 63.342(a)[1], which states:

> DOE's performance assessments conducted to show compliance with 63.311(a)(1), 63.321(b)(1), and 63.331 shall not include consideration of very unlikely features, events, or processes, i.e., those that are estimated to have less than one chance in 10,000 of occurring within 10,000 years of disposal.

There is, therefore, no contribution to the expected annual dose to the RMEI from criticality. This conclusion would be unchanged for any and all alternative analyses of criticality that result in a total probability of criticality less than 1 in 10,000 in 10,000 years.

As discussed in the responses to RAIs 3.2.2.1.2.1-4-035 and 3.2.2.1.2.1-4-040, the use of a subcritical margin does not directly impact the probability of postclosure criticality for commercial spent nuclear fuel (SNF) and DOE SNF. For commercial SNF, the use of a subcritical margin impacts the loading curve by increasing the number of assemblies in the unacceptable range of the loading curve. For DOE SNF, a subcritical margin beyond what has already been integrated into the analysis as presented in Table 1 of the response to RAI 3.2.2.1.2.1-4-035, impacts the design and loading requirements for DOE standardized SNF canisters. The postclosure criticality analysis established design and loading criteria, such that credible configurations within the regulatory period are subcritical.

2. COMMITMENTS TO NRC

None.

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[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.

3. DESCRIPTION OF PROPOSED LA CHANGE

None.

4. REFERENCES

None.

RAI: Volume 3, Chapter 2.2.1.2.1, Fourth Set, Number 43:

Provide information on how "unqualified" (i.e., with respect to the loading curve requirements) assemblies will be disposed of. Provide estimate of probability of criticality for the waste packages containing "unqualified" commercial assemblies accounting for degradation processes and manufacturing errors for any additional neutron absorbers to be used in these waste packages. Justify that the probability of criticality calculated for waste packages containing "unqualified" assemblies would not affect overall repository probability of criticality.

1. RESPONSE

Commercial spent nuclear fuel (SNF) burnup and enrichment are probable subjects of repository licensing specifications. As stated in SAR Section 2.2.1.4.1.1.3, transportation, aging, and disposal (TAD) canisters will be loaded with commercial SNF at the respective power facilities according to loading curves and shipped to the repository for aging (if needed) and subsequent disposal. Loading curves, which are functions of burnup and enrichment, are the loci of values delineating the region of acceptable burnup/enrichment combinations for postclosure criticality control. The loading curve is generated as part of the postclosure criticality analysis documented in SAR Section 2.2.1.4.1, and the assigned burnup values of all assemblies considered for loading into the TAD canister are compared directly against this loading curve. Assemblies having burnup values in the unacceptable range (i.e., below the minimum required burnup) must be loaded into canisters with additional reactivity control mechanisms (e.g., disposal control rod assemblies) (DOE 2008, Section 3.1.5(2)(a)(6)). Canisters loaded in this manner will be individually analyzed to show acceptable reactivity control for postclosure performance prior to receipt and acceptance at the repository. The details of the actual additional reactivity control mechanisms to be used, including consideration of reliability and neutron absorber material degradation processes, will be part of the individual canister analysis.

As stated in the response to RAI 4.2.5.1-1-009, prior to acceptance of commercial SNF, the DOE will perform reviews of commercial licensees and certificate holder quality-affecting work-related documentation 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 theDOE 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 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 administrative controls to be included in the license specifications required by 10 CFR 63.43. As described in SAR Section 5.10.2.4.2 and Table 5.10-3, these administrative controls will be compiled in the *Technical Requirements Manual*, controlled by 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. The

administrative controls described in SAR Section 5.10.2 and SAR Table 5.10-3 require completing similar analyses prior to receiving individual waste forms or canister/waste package design configurations not explicitly analyzed in the license application.

DOE plans to use the approach described above such that all SNF assemblies will be qualified waste forms prior to emplacement.

2. COMMITMENTS TO NRC

None.

3. DESCRIPTION OF PROPOSED LA CHANGE

None.

4. REFERENCES

DOE 2008. *Transportation, Aging and Disposal Canister System Performance Specification.* WMO-TADCS-000001, Rev. 1 ICN 01. Washington, D.C.: U.S. Department of Energy, Office of Civilian Radioactive Waste Management. ACC: DOC.20080331.0001.