UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

August 18, 1999

Dr. Stephen Brocoum
Assistant Manager for Licensing
U.S. Department of Energy
Office of Civilian Radioactive Waste Management
Yucca Mountain Site Characterization Office
P.O. Box 30307
North Las Vegas, Nevada 89036-0307

SUBJECT: U.S. NUCLEAR REGULATORY COMMISSION REQUEST FOR ADDITIONAL INFORMATION ON THE U.S. DEPARTMENT OF ENERGY TOPICAL REPORT ON DISPOSAL CRITICALITY ANALYSIS METHODOLOGY

By letter dated January 7, 1999, the U.S. Department of Energy (DOE) submitted the "Disposal Criticality Analysis Methodology Topical Report" for staff review. As indicated in our letter of February 11, 1999, the U.S. Nuclear Regulatory Commission (NRC) completed its acceptance review of this subject topical report and concluded that it satisfied the qualifications criteria for a topical report and was sufficiently complete for the staff to initiate its detailed technical review.

As a result of our ongoing detailed technical review of the topical report and information gained at the May 5, 1999, technical exchange on criticality, the staff has developed the enclosed request for additional information (RAI). The RAI contains a compilation of additional information requirements, identified todate by the NRC staff, during its review of the topical report. The Topical Report Review Plan, dated February 28, 1994, was used to review the topical report. Each individual question describes information needed by the staff to complete its review of the topical report and provides the technical basis for that request. In order for the staff to complete its review in accordance with the agreed upon schedule of May 20, 1999, the responses to the RAI should be provided within 30 days of the date of this transmittal letter.

As indicated in our telecon of May 20, 1999, the staff review of the topical report is limited to the disposal criticality analysis methodology and the planned approach to validating the methodology (e.g., appropriateness of using Commercial Reactor Criticals for criticality validation, or appropriateness of using non-repository environmental experiments for validating EQ3/6). Other areas are being reviewed only as needed to complete the review of the methodology and approach as noted above. The review is not covering the additional requests made in the topical report, such as approval of models (SCALE 4.3 with 44-energy group cross section, MCNP4B2, EQ3/6, etc.) or the specific benchmark experiments. If DOE requires NRC input on those aspects of the topical report outside the scope of the current NRC review, DOE should request this review in writing and a separate schedule will be developed to accommodate the review. In addition, the staff RAIs on the example application of the proposed methodology for commercial spent nuclear fuel (Appendix C) and DOE spent nuclear fuel (Appendix D) to the topical report should in no way be interpreted as acceptance of the methodology.

WM-11

NH16

102.2

The NRC staff is available to meet and discuss the RAIs at DOE's request. If you have any questions or comments on the above information or wish to schedule a meeting to discuss the RAIs, please contact Sandra Wastler, of my staff, at (301) 415-6724 or by e-mail at SLW1@nrc.gov.

Sincerely,

C. William Reamer, Chief
High-Level Waste and Performance
Assessment Branch
Division of Waste Management
Office of Nuclear Material Safety
and Safeguards

Enclosure: As stated

cc: See attached distribution list

Letter to S. Brocoum from C.W. Reamer dated: August 18, 1999

cc: R. Loux, State of Nevada
S. Frishman, State of Nevada
L. Barrett, DOE/Wash, DC
A. Brownstein, DOE/Wash, DC
S. Hanauer, DOE/Wash, DC
C. Einberg, DOE/Wash, DC
D. Shelor, DOE/Wash, DC
N. Slater, DOE/Wash, DC
R. Dyer, YMPO
R. Clark, YMPO
A. Gil, YMPO
S. Echols, M&O
B. Price, Nevada Legislative Committee
J. Meder, Nevada Legislative Counsel Bureau
D. Bechtel, Clark County, NV
E. von Tiesenhousen, Clark County, NV
J. Regan, Churchill County, NV
T. Cain, Esmeralda County, NV
L. Fiorenzi, Eureka County, NV
A. Remus, Inyo County, CA
T. Manzini, Lander County, NV
E. Culverwell, Lincoln County, NV
J. Wallis, Mineral County, NV
L. Bradshaw, Nye County, NV
M. Murphy, Nye County, NV
J. McKnight, Nye County, NV
N. Stellavato, Nye County, NV
D. Kolkman, White Pine County, NV
D. Weigel, GAO
W. Barnard, NWTRB
R. Holden, NCAI
A. Mitre, NIEC
R. Arnold, Pahrump County, NV
J. Lyznicky, AMA
R. Clark, EPA
F. Marcinowski, EPA
R. Anderson, NEI
R. McCullum, NEI
S. Kraft, NEI
J. Kessler, EPRI
G. McKnight, Pahrump, NV
R. Wallace, USGS
R. Craig, USGS
W. Booth, Engineering Svcs, LTD
S. Trubatch, Winston & Strawn

Request for Additional Information on DOE's
Disposal Criticality Analysis Methodology Topical Report
YMP/TR-004Q, Revision 0

CHAPTER 1.0 INTRODUCTION

1-1 Explain the basis for the following statement: "Present information is that HLW will not contain sufficient amounts of fissile material to pose a criticality risk, even in the absence of any criticality control material. Therefore, the only foreseen application of this analysis methodology to HLW will be to demonstrate this fact for a few worst-case configurations of moderator and geometry."

The sources for the "present information" are not provided. The validity of these statements can only be verified after the indicated information and demonstration analyses have been submitted.

Section 1.2 Objective

1-2 Explain why the Criticality Consequence criterion refers to consequences of only a single criticality event.

Certain classes of scenarios with common-mode or correlated pathways may lead to criticality of a number of packages over time with a probability that may or may not be much less than that of a single criticality in a single package. Once a certain scenario or pathway is established, criticalities in other similar packages by that pathway or a closely correlated one are not statistically independent. Therefore, in such cases one may need to consider the consequences (and probabilities) of more than one criticality event under this criterion.

1-3 Clarify the range of applicability of the methodology discussed in Item G.2, page 1-5.

The footnote on page 1-1 states that the methodology and processes are to be applicable to all different waste forms (WFs). Item G.2 is inconsistent with this in requesting consideration of the validation process for a limited class of waste forms (i.e., commercial Spent Nuclear Fuel (SNF)).

1-4 Explain why a broader range of configurations is not discussed in Item J.1, page 1-5.

The configuration identification process may fail to identify those configurations with the greatest potential consequences, i.e., configurations with potentially positive feedback. Such configurations should be identified using a process, supplementing the existing proposed method, whereby the most significant credible or postulated configurations are first identified and then either eliminated or further considered based on an evaluation of the probabilities of mechanisms that could produce such configurations.

Section 1.3 Scope

1-5 Explain why the scope of the TR does not correspond to the methodology and processes actually described in the TR.

The methodology and processes discussed in the Topical Report were developed primarily with commercial Light Water Reactor (LWR) fuel in mind. In addition, the LWR discussions are mostly restricted to Pressure Water Reactors (PWRs), with little consideration of the Boiling Water Reactors (BWRs). The staff expects to see numerous exceptions and differences in the methodologies ultimately used for naval Spent Nuclear Fuel (SNF), other U.S. Department of Energy (DOE) SNF, other highly-enriched materials, graphite-moderated fuel, and vitrified High-Level Waste (HLW). The applicability of the methodology described in the Topical Report to the broad variety of waste forms (e.g., all 250 DOE SNF types) cannot be established in this review.

Section 1.4 Quality Assurance

1-6 Clarify the statement that "the information presented in this topical report is not design information that can be used to support procurement, fabrication, or construction." with respect to Quality Assurance.

If the methodology is formulated (e.g., specifying critical limit, dismissing configurations not having potentials for criticality, etc.) based on the data presented in this report, and the design of the criticality control systems in the waste packages (WPs) is based on this methodology, it is not clear how these data are not used, directly or indirectly, in the design of the waste package. In particular, some of the references (e.g., CRWM M&O 1998e) state that "this document will not directly support any DOE Office of Civilian Radioactive Waste Management (OCRWM) construction, fabrication, or procurement activity and therefore is not required to be procedurally controlled as to be verified (TBV)."

1-7 Specify what part of the Actinide-Only Burnup Credit Topical Report is used in this topical report.

The "Topical Report on Actinide-Only Burnup Credit for PWR Spent Nuclear Fuel Packages (DOE 1997)" has not been approved by the NRC Spent Fuel Project Office.

Section 1.5 Overview of the Methodology

1-8 Justify the approach used to dismiss configuration classes as having the potential for criticality based on an evaluation of a given configuration with parameter values at some selected points for each of the configurations (Figure 1-1).

If configuration classes are dismissed based on evaluation of a given configuration with parameters values at several points rather than examination of full range of parameter values, it is possible to dismiss a configuration or even configuration class which has potential for criticality. It seems it would be more appropriate to perform the criticality analyses for a full range of parameter values for each configuration, examine those

results against the Critical Limit criterion, and then decide if that particular configuration or configuration class is acceptable for disposal from a criticality standpoint.

CHAPTER 2.0 REGULATORY PERSPECTIVE

2-1 Throughout this section, there are references to Regulatory Guide 3.58. However, it, as well as 3.1, 3.4, 3.43, 3.45, 3.47, 3.57, 3.68, 3.70, and 8.12, has been superseded by Regulatory Guide 3.71, published in August, 1998. The references should either be updated or an explanation for the choice to use Regulatory Guide 3.58 should be provided.

CHAPTER 3.0 METHODOLOGY

Section 3.1 Standard Criticality Scenarios

3-1 Indicate how the effects of disruptive events will be considered in the evaluation of potential criticality events in the repository.

The scenarios listed in Figures 3-1 and 3-2 appear to be comprehensive for an undisturbed repository. However, it is not clear whether the potential effects of disruptive events have been adequately considered. Failure to consider all potential scenarios that could result in a criticality event could result in an underestimation in the probability of a critical event occurring within the repository. Some potential effects of disruptive events include:

(a) Seismic events could cause the waste packages to rotate on their invert, potentially allowing corrosion products to be released while the hole in the waste package is facing down. Later, the hole could rotate back to the top of the package, allowing the package to fill with water.

(b) A volcanic event, although a low probability event, could fail many waste packages and force their contents into a compact configuration encased in lava at the end of the tunnel.

3-2 Justify the scenario selection process used to focus on the degradation modes and phenomena that produce the critical configurations of interest.

The process as now proposed may not identify and address some scenarios that should be considered. For example:

(a) The internal scenarios do not give adequate consideration to criticality at the less-burned ends of the fuel. Even in scenarios where the basket poison material, or absence thereof, is uniformly distributed over the length of the active fuel material, criticality will occur predominantly at the fuel ends. End-effect criticality is made even more relevant by scenarios where fuel and basket poison material are displaced axially relative to one another. Such displacement or shifting would remove poison from where it is most needed, i.e., at the ends.

(b) For external near-field and far-field criticality, more focused consideration should be given to scenarios with potentially positive neutronic feedback characteristics. This could be done by first identifying hypothetical configurations that produce positive feedback effects and then evaluating the credibility or likelihood of mechanisms that might form such configurations. The methodology as now proposed does not seek and address those criticality scenarios that have the greatest potential consequences. Furthermore, the probability criterion for events with potentially high consequences should be lower than that for events with lesser consequences.

3-3 Explain why a discussion of the fast-fissionable, non-fissile actinides that by themselves can sustain a critical chain reaction is not included in this section.

The discussion of fast criticality scenarios in which little or no moderation is required should be extended to include "minor actinides" (see ANSI/ANS 8.15) that by themselves can sustain a critical chain reaction with fast neutrons only. Such actinides are sometimes called "fissible." The TR should indicate how these actinides have been considered with regard to their abundance over time in various waste forms and should discuss the bases for any conclusions about their significance (or lack thereof) to repository criticality.

3.1.2 External Scenarios

3-4 Confirm that far-field configuration classes FF-3c, 3d, and 3e are located in the saturated zone.

The text in item 1 of section 3.1.2 contradicts figure 3-2b in assigning these configurations to the unsaturated zone. The distinction is important in modeling hydrologic and geochemical processes.

3-5 Explain why, in item 3, configuration NF-1b includes only a reducing reaction with tuff as a mechanism for precipitation of fissile solutes in the near-field below the waste package.

Other chemical reactions should be considered as causing such precipitation, such as changes in aqueous chemistry related to the presence of concrete and tuff. This comment reflects the desire for completeness in modeling the configurations.

3-6 Clarify whether credit will be taken in the criticality analyses for the assumption that there is no mechanism for completely sealing the fractures in the bottom of the drift so any in-drift accumulations of water will only be present for a few weeks.

Previous investigations have indicated that thermal alteration of the rock surrounding the repository or microbial growth has the potential to seal fractures, at least in local portions of the repository (Lin and Daily, 1989[1]). Although it appears that the scenarios

[1]Lin, W., and W.D. Daily. 1989. *Laboratory Study of Fracture Healing in Topopah Spring Tuff–Implications for Near-Field Hydrology*. UCRL–100624. Livermore, CA: Lawrence Livermore National Laboratory.

listed in figure 3-2 include the potential for water to pond on the bottom of the drift, if credit is taken for the short duration of ponding water in the drift, this assertion that fractures cannot become sealed will have to be justified.

3-7 Explain why, in item 3, configuration NF-1b includes only a reducing reaction with tuff as a mechanism for precipitation of fissile solutes in the near-field below the waste package.

Other chemical reactions should be considered as causing such precipitation, such as changes in aqueous chemistry related to the presence of concrete and tuff. This comment reflects the desire for completeness in modeling the configurations.

3-8 Correct item 5 to state that the final two configurations are NF-3b and 3c, rather than FF-3b and 3c.

The context of the sentence implies incorrectly that the listed sites of colloidal accumulation are all in the far field. In addition, in the final sentence "open fractures" should be specified as being in concrete to be consistent with Figure 3-2a. These changes will correct the impression from item 5 that all colloidal accumulation sites are far-field.

Section 3.2 Determining Internal Configurations

3-9 State whether temperature is included among the parameters quantified at this stage of the methodology and describe possible thermal variations.

Because equilibrium states and degradation/reaction rates for WP internal components (including WF) are temperature-dependent, all geochemical modeling should include sensitivity to temperature variations, including those caused by repository heating and cooling and by criticality events. With respect to internal configurations, the example analysis of appendix C refers to EQ6 calculations described in CRWM M&O (1998e[2], appendix C, reference list). The discussion in this reference does not explicitly mention temperature constraints on models. Recent DOE modeling of the near-field (Harbin, 1998[3]) predicts that temperatures of close to 100 °C may persist at the repository horizon 5000 yr after closure. A more recent repository design, EA.-II[4] yields lower drift temperatures, but the waste package would still experience temperatures above 60 °C for at least 2000 yr. Thermal variations have a strong effect on degradation processes and rates for Waste package internal components. Furthermore, high temperatures would affect water chemistry [e.g., see composition of J-13 equilibrated with tuff at 90

[2] CRWM M&O. 1998e. *EQ6 Calculation for Chemical Degradation of PWR LEU and PWR BOX Spent Fuel Waste Packages*. BBA000000–01717–0210–00009. Revision 00. Las Vegas, NV: CRWM M&AMBLE.19980701.0483.

[3] Hardin, EL 1998. Near-Field/Altered Zone Models Report. UCRL–ID–129179. Livermore, CA: Lawrence Livermore National Laboratory.

[4] Snell, R. 1999. Presentation to the Advisory Committee on Nuclear Waste.

°C in Wronkiewicz et al. (1992)[5]. This comment applies also to discussions of internal and external geochemistry models in topical report sections 3.3, 4.2.2, and 4.2.3.

Section 3.4 Criticality Evaluation of Configurations

3-10 Justify the use of the fresh fuel assumption in the internal criticality evaluations for waste forms other than commercial and naval SNF.

Many types of fuel that contain burnable poisons can be more reactive at moderate levels of burnup than when fresh. For such fuels, analysis of poison depletion and other burnup reactivity effects may be needed, not for burnup credit, but rather as a way of bounding the potential in-package burnup "debit."

Section 3.4.1 Computer Codes

3-11 Justify the use of the fresh fuel assumption in the external criticality evaluations for waste forms other than commercial and naval SNF.

Criticality evaluations for near-field and far-field configurations must consider the actual compositions of SNF materials. Using the fresh fuel composition would not be bounding for scenarios where uranium, plutonium, and other fissionable actinides have different potentials for mobilization and reconcentration.

Section 3.4.2 Material Composition of Commercial SNF

3-12 Explain how the neutron-induced breeding of fissile and fissionable nuclides over time in the repository has been evaluated.

The TR does not indicate whether breeding effects have been evaluated. A scoping analyses of neutron sources, including (α,n) reactions, and associated breeding reactions should be provided or referenced in the TR. The evaluation should consider all fertile nuclides present in the various waste forms (e.g., U-238, Th-232, Pu-240). This RAI also applies to the material in section 3.4.3.

Section 3.4.3 Principal Isotopes for Commercial SNF Burnup Credit

3-13 Explain why the verifiability of the inventory, as part of isotopic validation, is not one of the criteria considered in selecting the principal isotopes for burnup credit.

As indicated in the report, nuclear, physical, and chemical properties of neutron-absorbing isotopes were considered in selecting them for burnup credit. The verifiability of isotopes for pre-closure configuration, in terms of isotopic validation, should also be one of the criteria in selecting the isotopes which can be used in subsequent post-closure isotopic inventory for criticality calculations.

---

[5]Wronkiewicz, D.J., J.K. Bates, T.J. Gerding, E. Veleckis, and B.S. Tani. 1992. Uranium release and secondary phase formation during unsaturated testing of UO_2 at 90°C. *Journal of Nuclear Materials* 190: 107–127.

3-14 Provide isotopic importance as a function of time which includes decay and loss of isotopes from spent fuel degradation in the "*Principal Isotope Selection Report* (CRWMS M&O 1998f) , and provide assurance for the presence of these isotopes within the spent fuel matrix for criticality control given the degradation of the spent fuel assemblies.

The degraded configuration described in CRWMS M&O 1998f indicates intrusion of water into the fuel rods without considering the reduction of isotopes through their dissolution in the water.

3-15 Justify taking credit for the isotopes in Table 3-1 through verification of their quantities predicted by the isotopic model.

To assume that the spent fuel is composed of the 29 isotopes listed in Table 3-1, one must verify the quantity of the isotopes predicted by the isotopic models. The isotopic models predict the radionuclide inventory as the function of reactor operating history. This validation must be performed by direct comparison of calculated to the measured isotopic inventory. Under-prediction or over-prediction of an isotope will have a direct effect on predicting the criticality potentials of a waste package accurately.

Section 3.5 Estimating Probability of Critical Configurations

3-16 Answer the following questions related to computational feasibility and the discussion on page 3-21:

(a) How many repetitions, or histories, are envisioned for the Monte Carlo simulation?
(b) What is the confidence limit on a calculated criticality probability?
(c) How are the probabilities for different configuration classes combined?
(d) What will be done if one of the intermediate steps cannot materialize (i.e., a plausible regression form cannot be obtained)?

In order to establish a 10^{-4} probability, a very large number of simulation runs must be generated. Additionally, in calculating criticality probabilities, it is desirable to calculate the associated confidence limit(s) about that probability. Answers to these questions could provide a clearer picture of the described methodology for estimating the probability of critical configurations.

3-17 Justify the assumption that Fe_2O_3 is the product that is formed by the corrosion of iron.

Credit is being taken for the filling of breached WPs by iron corrosion products, namely Fe_2O_3, thereby limiting the quantity of water present. It is unclear why the possibility that some of the iron corrosion product may be in the form of FeOOH was not considered. Justification of why the formation of FeOOH in lieu of Fe_2O_3 was not considered should be provided or else the effects of FeOOH formation on criticality control should be determined.

3-18 Indicate whether the criticality calculations will account for neutron interactions between WPs.

The calculation of the effective neutron multiplication factor, k_{eff}, should account for all fissile material that can impact the modeled system. The EDA-II[6] design places the WPs much closer together in the line-loading formation, which will lead to greater neutronic interaction between the packages. It is not clear from the topical report whether these effects will be accounted for when calculating the k_{eff} of the fuel both inside the WP and in the near-field.

Section 3.6.1 Type of Criticality Event

Slow versus fast reactivity insertion rate

3-19 Explain why the configuration with seismic event causing reshuffling of the spent fuel and the spent fuel being fully submerged in the water inside the waste package is not considered as a plausible scenario for the fast reactivity insertion rate.

The reference cited in the topical report (CRWMS M&O 1997a) provides earthquake consequence analysis with respect to criticality in terms of iron oxide settling in the bottom of the waste package and providing the transient criticality analysis.

With respect to iron oxide, it has not been demonstrated that the iron oxide can remain in the waste package. Secondly, the reshuffling of the spent fuel assemblies during a seismic event is a more plausible scenario than the iron oxide mixing with water and becoming a homogenous solution. Thirdly, if even the iron oxide would remain in the waste package, the settled iron oxide configuration is the initial condition and the uniformly distributed configuration is the condition right after the seismic event. It is not clear how the scenario is postulated with these two conditions being reversed. Therefore, the reshuffling of spent fuel in the time frame of a second or less without the iron oxide is the more realistic scenario than the one presented in CRWMS M&O 1997a.

Steady-state versus transient

3-20 Explain why, for transient criticalities, you cannot have conditions 1 and 2 met under a more realistic scenario with seismic event and partially-flooded waste package with no iron oxide.

The confining condition stated in the topical report is not needed for an impact of fast reactivity insertion in the waste form. With respect to the second condition, the fast reactivity insertion is plausible under the seismic event with the top row assemblies rolling over and being submerged inside the waste package or fissile materials reshuffling and coming together outside of the waste package. With regard to the third condition, even for the optimistic condition described in the topical report, the k_{eff} for inside the package is 1.0189 which is super prompt criticality. Therefore, the rate of energy release is very fast.

[6]Snell, R. 1999. Presentation to the Advisory Committee on Nuclear Waste.

Under-moderated versus over-moderated

3-21 Justify why moderation was the only mechanism used to govern the positive or negative feedback characteristics of a critical system.

The topical report's current discussion does not recognize, for example, that in certain configurations water is a poison and that other moderators (SiO_2) more strongly influence the thermal neutron spectrum. Furthermore, particle self-shielding mechanisms for absorbers and fissile materials can have important implications not normally associated with the concept of over/under-moderation. Reflection dynamics likewise may be important in certain scenarios.

The concept of under/over-moderation has limited applicability outside LWR cores. For example, the 1986 Chernobyl disaster, by far the worst autocatalytic criticality event in history, was governed by positive void reactivity effects that have nothing to do with the concept of over-moderation. The positive void reactivity effects in CANDU reactors are likewise unrelated to over-moderation.

Especially in configurations where positive feedback effects are deemed credible, it is important to analyze the dynamic progression of criticality events using appropriately coupled models of the actual neutronic and thermal-mechanical phenomena in that system. Repository physics can differ fundamentally from LWR core physics. Correct analysis of the criticality dynamics is essential to assessing any potentially disruptive effects in the repository.

3.6.2 Evaluating Direct Criticality Event Consequences

3-22 Justify the statement that accumulation and geometry of fissionable mass needed for large disruptive criticality events is expected to be beyond anything physically possible in the repository.

It is not clear why the statement is made when another statement in the same paragraph explains that "some theoretical analyses have identified larger, disruptive consequences . . . "

Section 3.7 ESTIMATING CRITICALITY RISK

3-23 Justify the assumption that the only detrimental effect of a criticality event on the repository performance is the generation of additional radionuclide inventory.

In addition to the increase in radionuclide inventory, other direct and/or indirect potential criticality consequences must be considered. Increase in the waste package heat output affects the near-field environment and the rate of material corrosion and waste form degradation within the waste package. Additionally, large disruptive criticality transients could generate sufficient heat and pressure to degrade the waste package, cladding, or spent fuel. This degradation of the waste form could increase the release rate of radionuclides and the corresponding dose at the critical group location. This comment

also applies to Section 1.2, Section 4.4.1.2, Section 4.4.1.1, Section 4.4.1.2, and Section 4.5.

Chapter 4.0 MODEL DESCRIPTION

Sections 4.1.1.2 Postclosure Isotopic Concentrations

4-1 Explain how the so-called bounding bias and uncertainty values are derived from a stochastic process.

For example, are statistical confidence intervals associated with the uncertainty values? Are these a function of the number of Monte Carlo histories? In which way is the Monte Carlo analysis used to derive these values of bias and uncertainties?

Section 4.1.3 Neutronic Model Validation

4-2 Justify the applicability of Commercial Reactor Criticals (CRC) for validation of MCNP4B in light of the lack of cross section libraries as a function of temperature.

It is not clear how well the MCNP4B cross sections can be validated against CRC when the modeling of CRC requires codes with cross sections as a function of temperature. MCNP4B dose not have the required cross section libraries as a function of temperature.

Section 4.1.3.1.3 Radiochemical Assays

4-3 Provide information on the initial enrichments and burnup for the new Radiochemical assay measurements that are being conducted to supplement the data base for commercial SNF isotopic model validation.

Staff notes that the existing data are limited to enrichments between 2.45 and 3.87 wt% ^{235}U. The new data should be for higher initial enrichments and higher burnup.

Section 4.1.3.1.4 Requirements for Isotopic Model Validation

4-4 Justify the use of 45 reactor core state points to bound the spent fuel operating history parameter values of the historical and projected spent fuel discharge for the spent fuel assemblies which are destined for disposal in the proposed repository.

The operating history parameter values of the 45 reactor core state points do not bound the operating history parameter values of the 100,000 or so commercial spent fuel assemblies which will be placed in the proposed repository at Yucca Mountain. The bounding operating history parameter values must be established based on the operating history parameter values of the historical and projected spent fuel assemblies to be discharged from the reactors.

4-5 Justify the method used to determine the isotopic code bias.

The purpose of code validation is to quantify the bias and the uncertainty which may exist within the isotopic code. The main problems with the approach described in 4.1.3.1.4 are:

(a) Not using ANSI/ANS 8.1 and 8.17 to establish area and range of applicability.

(b) Using the CRC operating history, which is insufficient to cover the complete range of operating history parameters of the discharged PWR spent fuel assemblies destined for disposal, to establish the bounding parameter values.

(c) Using the integral k_{eff} approach, which takes advantage of compensating errors in isotopic prediction to validate the isotopic model.

(d) Using the established parameter values from Part a to perform calculation-to-calculation comparison, as opposed to comparing calculations to experimental results for the purpose of isotopic model validation.

Other approaches such as direct comparison of measured to calculated values would eliminate some of these concerns.

4-6 Explain why k_{eff} adjustment approach, which takes advantage of compensating errors in isotopic inventory, is chosen over the direct adjustment of each isotopic inventory for capturing the isotopic decay and branching ratio uncertainties.

Section 4.1.3.2 Determination of Critical Limits

4-7 Provide a justification for not incorporating the following information into Figure 4-1 for estimating Critical Limit.

(a) Identification of subsets of validation experiments which are applicable to the waste form and the configuration classes within and outside the waste package.

(b) Performance of a normality test prior to applying any of the statistical analyses such as regression analysis, which is based on the normality assumption. Figure 4-1 shows that the normality test is performed after the regression analysis indicates there is no trends. The base assumption for regression analysis is normality which must be verified through some statistical tests.

(c) Performance of a regression fit of k_{eff} on predictor variables for the **relevant subset** to identify trending parameters.

(d) Inclusion of all the parameters, not just the ones with "strongest correlation," which have statistically significant trends as the function of k_{eff}.

4-8 Provide the technical bases (other than "commonly used") for using 0.05, instead of 0.01 or 0.001, for the level of significance in identifying linear trends with respect to the trending parameters.

Although it is indicated that approval of a specific value for the level of statistical significance will be sought in the License Application, the TR should provide a statistical rationale used for selecting the specific value.

4-9 Justify the basis for redefining Δk_m.

ANSI/ANS-8.17 defines Δk_m as "an arbitrary margin to ensure the subcriticality k_s." The examples provided for Δk_m in the topical report such as "1) the effect on k_{eff} associated with the long-term decay of radionuclides in the waste form and 2) the effect on k_{eff} associated with extending the range of applicability of the CL beyond the experimental database" are the standard biases which must be included as part of isotopic bias and $\Delta k_c(x)$, respectively. The Δk_m in this case must include a subcritical margin. For example, if CL for a particular configuration is established to be 0.95, is the value of 0.9499 for $k_s + \Delta k_s$ considered to be subcritical? If more neutron histories are used, the calculated value could be 0.95 or beyond. Therefore, the need to identify a zone of criticality and incorporate it into the total uncertainty should be considered.

This comment also applies to Normal Distribution Tolerance Limits (Section 4.1.3.2.2) and the Distribution Free Tolerance Limit (Section 4.1.3.2.3).

4-10 Justify the use of the linear regression model to fit the data presented in Figure 4-2.

Considering the data in Figure 4-2, it seems that another model, i.e., exponential or polynomial, could better fit the data than the proposed linear regression model for trending criticality level.

4-11 Justify why a single predictor is used for the least-square fits, as explained in the discussions. Examine the data to determine whether a combination of factors would yield a better fit.

One could argue that a "less sensitive" model (a model that does not include all significant factors and factor combinations, or a model with a nonlinear structure) is more conservative. This argument would be correct with respect to the measure of uncertainty, since a poor fit is associated with larger uncertainty. However, a more refined regression could have a negative trend that may be undetected due to the simplicity of the model. Therefore, the question is whether the failure to detect a negative trend is outweighed by the large measure of uncertainty.

4-12 Explain how parameters other than those used for trending are applied to characterize a system and the benchmark experiments.

The extension of the range of applicability (ROA) must be addressed with caution. How would one know of any trending effect outside the experimental ROA?

Section 4.1.3.2.1 Lower Uniform Tolerance Band

4-13 Justify why the application of combined Method 1 and 2 in Lichtenwalter et al(1997, pp. 158-162) as referenced in the topical report was not evaluated.

As stated in Lichtenwalter et al., "the recommended purpose of method 2," Lower Uniform Tolerance Band (LUTB), "is to apply it in tandem with Method 1..." The term Δk_m must be included in the LUTB approach. However, the value for Δk_m may be determined based on some reasoning as opposed to the traditional 5% administrative margin.

Section 4.1.3.2.2 Normal Distribution Tolerance Limits

4-14 Clarify if the Normal Distribution Tolerance Limits (NDTL) are based on the prediction interval or tolerance interval, and justify this approach.

The prediction interval is based on predicting, with a pre-determined confidence level, a single future value which would be below the critical limit. On the other hand, tolerance limit predicts a percentage of future values which would fall below the critical limit. The latter is a more acceptable approach.

4-15 Justify elimination of Δk_m in the Critical Limit for Normal Distribution Tolerance Limits (NDTL).

Same argument provided for LUTB with respect to Δk_m can be applied to NDTL.

Section 4.1.3.2.3 Distribution Free Tolerance Limit

4-16 Demonstrate that the Distribution Free Tolerance Limit (DFTL) approach is at least as bounding as the lowest-k_{eff} approach.

It appears that the selection of the Ith k_{eff}, which is based on the I number of samples needed to provide the desired tolerance limit (e.g., 95/95), does not result in a low k_{eff} value for the Critical Limit. For example, based on the explanation provided by Natrella (1966, pp. 2-15), referenced in the topical report, using 95 critical benchmarks and 95 % confidence level, k_{eff} for 95% of waste packages under a specific configuration for specific waste type will be below the third largest k_{eff} for the 95 critical benchmarks. What is needed is that with 99% confidence level, 95% of population (e.g., k_{eff}) fall below the smallest k_{eff} for the benchmark set.

4-17 Justify elimination of Δk_m in the DFTL approach.

Same argument provided for LUTB with respect to Δk_m can be applied to DFTL.

4-18 Explain the use of the "3 standard deviations (3σ)" limit in a distribution-free mode.

It is not clear why the "3 standard deviations (3σ)" is used. Is the 3σ enough to capture all possible scenarios?

Section 4.1.3.3.1 Range of Neutronic Parameters

4-19 Provide a justification for not using a systematic approach used to identify the area and range of applicability with respect to criticality model validation for each configuration class and waste form.

The approach outlined in Section 4.1.3.3.1 is neither fully consistent with the approach in Lichtenwalter et al., nor is it comprehensive and complete with respect to identifying those parameters which may exhibit a trend in the criticality code bias.

Material concentrations, geometry, and spectrum are the areas (i.e., area of applicability (AOA)) within which the benchmarks must be evaluated for their applicability to the specific configuration class and waste form. Furthermore, there are sub-areas, if you will, within each of these AOAs which categorize the substantial variances within each of these AOAs, some of which are indicated in Page 4-18. Then, subsets of benchmarks which are based on waste package configuration class, waste form, and/or benchmark classes (e.g., Table 4.1 in Lichtenwalter et al.) need to be identified. After that, specific variables which can represent each of those categories and presence or absence of any associated statistically significant trends must be identified.

4-20 Clarify why the values for AENCF in CRWMS M&O 1998n are in mega electron volt (MEV) range as opposed to fractional or single digit electron volt.

Staff notes that AENCF inappropriately weights higher energy neutrons, resulting in thermal systems having an AENCF in the 10 keV range, whereas the predominant fission rate spectrum is actually centered in the 0.1 eV range of neutron energies. Use of Energy of Average Lethargy of neutrons causing Fission (EALF) will correct this problem. SCALE4.4 now includes the EALF parameter in its output. A corresponding Type 4 tally specification for MCNP4B can be designed by the code user.

Section 4.1.3.3.3 Extension of the Range of Applicability

4-21 Provide the rationale for switching from LUTB method to NDTL method for extending the range of applicability.

The method for determining Δk_c must be based on the 99.5% of future calculations as opposed to single future calculation on which NDTL may be based. Furthermore, the margin or zone of criticality must be included in Δk_m.

4-22 Describe the approach in establishing an additional margin when performing extrapolation beyond the range of applicability.

The report indicates that an additional margin will be added when extrapolation is extended beyond the range of applicability. However, it does not discuss the approach in establishing or quantifying this additional margin. Discussion with regard to the approach in establishing additional margin beyond the range of applicability is needed.

4.1.3.4 Discussion of Results

4-23 Present the results in terms of their applicability to the waste package configurations under repository conditions with respect to material, geometry, and spectrum.

Table 4-1 presents only the results of modeling and calculating k_{eff} for the Laboratory Critical Experiments (LCE) and CRC without making any connection to their applicability to the different waste package configurations in the repository with respect to specific ranges of parameters covering material, geometry, and spectrum.

Section 4.1.3.4.1 Trending Results for Commercial Spent Nuclear Fuel

4-24 Demonstrate the applicability of CRCs to the waste package configuration with the intact waste form with respect to the following areas:

a) Material (e.g., plate boron concentration, soluble boron concentration, reflector composition, fuel material properties, etc...)

b) Geometry (e.g., assembly separation distance, poison plate thickness, reflector wall separation distance, etc...)

c) Spectrum (e.g., Average Energy for Neutron Causing Fission (AENCF) compared to AENCF for intact waste form)

CRWMS M&O 1998n does not establish the applicability of CRCs to the waste package with the intact waste form as requested in this topical report. For example, Table 2.4-1 on Page 60 of CRWMS M&O 1998n shows the AENCF range for CRCs are only between 0.2475 MEV and 0.2643 MEV. However, the same table shows the AENCF range for all the configurations in the repository is between 0.0016 MEV and 0.3311 MEV. Assuming the AENCF range for the waste package configuration with intact spent fuel assemblies is somewhere between 0.0016 and 0.3311 (the report should specify the AENCF along with all the relevant benchmarking parameter ranges for the intact spent fuel assemblies), at least the CRC range with respect to AENCF spectral index must cover the waste package configuration with the intact waste form.

Section 4.2.1 Corrosion Models

4-25 Justify the extensive reliance on the wide range of corrosion rates utilized to determine the probability and location of a WP breach.

Given sufficient criticality control in the as-fabricated WP, a breach in the WP is necessary for a criticality event to occur. The model used to determine the probability of a WP breach and its location was the WAPDEG code using the Total System Performance Assessment (TSPA)-Viability Assessment (VA) base case. The primary limitation of this case is that the input parameters for corrosion rate rely extensively on expert elicitation, with nearly five orders of magnitude variance in the corrosion rate utilized. Thus, the possibility exists for a wide range of WP failure times and a commensurately wide range of times in which criticality control becomes important. The

wide range of corrosion rates resulting from the heavy reliance on expert elicitation is considered a limitation to the utility and validity of the subsequent criticality analysis, since it leads to dilution of the probability of occurrence and resulting risks.

4-26 Provide justification for the long-term credit being taken for the presence of fuel cladding in the degradation analysis.

It appears that credit is being taken for the presence of Zircaloy-4 cladding in terms of its corrosion resistance. Although Zircaloy-4 does have good corrosion resistance, it is known to suffer from localized corrosion under reasonably attainable conditions inside breached WPs. Additionally, the cladding can be degraded prior to disposal due to the effects of irradiation, reactor water chemistry, and predisposal storage conditions. Commercial SNF exhibits a wide range of Zircalloy material characteristics, including large variations in the degree of hydriding, oxidation, erosion thinning, embrittlement, crack formation, pellet-cladding interactions, crud depositions. Further information is requested on the technical basis that this degree of credit can be claimed for the Zircaloy-4 cladding and the effect on criticality control if no credit is taken.

Section 4.2.2 Internal Geochemistry Models

4-27 Clarify the internal geochemistry model treatment of Uranium (U) produced from WF degradation.

WF U is input into solution (along with other degradation products) according to fixed degradation rates and solution evolution modeled in EQ6. What is not clear is if U secondary phases are allowed to precipitate, in effect lowering the U release rate and perhaps lowering the probability for potentially critical external accumulations. References cited in the topical report suggest that secondary U phases will be included in internal degradation models. These references include CRWMS M&O (1998e)[7] and CRWMS M&O (1998q)[8], the latter being cited in the former. For example, in section 6.3.2 of CRWMS M&O (1998e), EQ6 models of SNF degradation are said to lead to precipitation of the hydrated uranyl silicate soddyite. In contrast, retention by secondary U phases was not modeled in the TSPA-VA. Clarify whether these differing approaches will be reconciled in future work.

4-28 Compare and contrast the approach to modeling to be used for release and internal geochemistry in the criticality analysis with that employed in present and future TSPA models (which may or may not use EQ3/6).

It was stated in topical report section 4.2.1 that WF degradation modeling in the criticality analysis will employ TSPA model approaches, but it is not clear if this extends

---

[7] CRWMS M&O. 1998e. *Evaluation of Codisposal Viability for Aluminum-Clad DOE-Owned Spent Fuel: Phase II Degraded Codisposal Waste Package Internal Criticality.* BBA000000–01717–5705–00017. Revision 01. Las Vegas, NV: CRWMS M&O.

[8] CRWMS M&O. 1998q. *Geochemical and Physical Analysis of Degradation Modes of HEU in a Codisposal Waste Package with HLW Canisters.* BBA000000–01717–0200–00059. Revision 01. Las Vegas, NV: CRWMS M&O.

to modeling release from WPs. TSPA-VA did not explicitly employ EQ3/6 geochemical modeling of WF alteration, RN release, and secondary solid phase formation. Any deviations from the TSPA-VA approach in the criticality analysis should be demonstrably more conservative or supportable.

4-29 Specify what kinetic models will be used in the internal geochemistry models and clarify whether default EQ6 values will be used.

Selection of kinetic models profoundly influences model results regarding degradation products and water chemistry. Cited documents discussing EQ6 degradation models (CRWMS M&O, 1998e, 1998q, and Appendix C reference 1998e) do not address kinetic models affecting the rates at which WP and WF degradation products precipitate. It appears that either default EQ6 kinetic parameters are utilized or kinetics are not included. Because degradation products are integral to criticality models, calculations predicting their formation should rely on supportable or conservative kinetic data. This comment applies also to the external geochemistry models discussed in section 4.2.3 of the topical report.

4-30 Justify the use of J-13 well water as representative of the solution that would be present within the WP.

Once the WP is breached, corrosion and degradation of WP internals play important roles in criticality control. Although some testing data has been obtained, the results are based on experiments conducted in variants of simulated J-13 well water. It seems unlikely that the chemistry of the solution inside a WP would be J-13. Furthermore, it is unclear if the testing program and the subsequent analysis has considered the possibility of chemistry changes resulting from evaporative processes and dissolution products from WP components (e.g., acidification due to metal cation hydrolysis, alkalinization from dissolution of HLW glass, etc.). Because of possible chemistry changes, the corrosion mode and corrosion rates could be altered from the general corrosion case considered. For example, alkalinization could lead to the formation of a passive film on carbon steel components that could then experience localized corrosion in the form of pitting or crevice corrosion in the presence of chloride. Similarly, the corrosion mode of stainless steel components could change from relatively slow passive dissolution to more rapid localized corrosion, which could lead to unanticipated, catastrophic failure of WP internal components. These accelerated corrosion modes could make conditions for criticality more favorable by allowing fuel materials to coalesce. Further information justifying the environments chosen and any further work examining likely alternate chemistries and their effects on material degradation is requested.

Section 4.2.3 External Geochemistry Models

4-31 Describe how colloidal deposition will be incorporated into modeled chemical deposition.

Indicate whether the approach will be the same as those adopted under TSPA. Evaluation of models of fissile material accumulation requires full understanding of colloid modeling. A previous analysis of external criticality (CRWMS M&O, 1998p)

concluded that colloidal transport and accumulation of fissile materials would be insignificant. It should be clear how new analyses will differ and to what extent they are supportable and conservative.

Section 4.2.4.1 Validation of Degradation Methodology

4-32 Provide more information on validation methods for the "pseudo flow-through" internal and "open system" external EQ6 models.

With regard to the "pseudo flow-through" model, topical report section 4.2.4.1 refers only to hand calculations supporting the solute concentration adjustments (CRWMS M&O, 1998q). This exercise only partially addresses the question of the validity of the model results.

In discussing the "open system model," it is stated that the results are conservative, but the pertinent reference (CRWMS M&O, 1997f) is missing from the chapter 6 reference list. The report acknowledges that validation has not yet been done, but does not describe how it will be done. This information is vital to assessing the methodology (see also discussion of topical report section 4.2.4.2 below). Validation approaches should provide confidence that models will not underestimate the effects of processes that could lead to criticality.

Section 4.2.4.2 Validation of the EQ3/6 Geochemistry Code

4-33 Provide additional information on the validation of EQ3/6 for the specific applications. The validation examples provided in topical report section 4.2.4.2 (Bourcier, 1994[9]; Bruton and Shaw, 1988[10]; Bruton, 1996[11]; Wolery and Daveler,1992[12]) do not adequately cover the conditions and processes to be included in the models. For example, the validated spent fuel and HLW models (Table 4-3) did not include the other waste package components (e.g., metal plates) to be included in the internal models. In addition, no examples are given that are comparable to the external models of low-temperature interaction between drift effluent waters and fracture walls. The DOE should state whether or not any new analyses will be performed that would support validation under the conditions to be modeled and, if not, how model confidence will be improved.

---

[9] Bourcier, W.L. 1994. *Critical Review of Glass Performance Modeling.* ANL9417. Argonne, IL: Argonne National Laboratory.

[10] Bruton, C.J., and H.F. Shaw. 1988. *Geochemical Simulation of Reaction Between Spent Fuel Waste Form and J-13 Water at 25 °C and 90 °C.* ISBN 0931837–80–4. Pittsburgh, PA: Materials Research Society.

[11] Bruton, C.J. 1996. *Near-Field and Altered-Zone Environment Report, Volume II.* Livermore, CA: Lawrence Livermore National Laboratory.

[12] Wolery, T.J., and S.A. Daveler. 1992. *EQ6, a Computer Program for Reaction Path Modeling of Aqueous Geochemical Systems: Theoretical Manual, User's Guide, and Related Documentation (Version 7.0).* UCRL–MA–110662 PT IV. Livermore, CA: Lawrence Livermore National Laboratory.

4-34 Provide additional information on the validity or conservatism of geochemical parameters to be used in EQ3/6 models.

As acknowledged in the report, there are large uncertainties in thermodynamic and kinetic data used by EQ3/6. The report states that a range of reaction rate values will be used so that conservative cases may be identified. This analysis should take account of any synergistic effects of varying rates for the numerous solid phases involved in this complex system. Such analysis should also be applied to thermodynamic data, particularly with regard to actinide phases. (Note, for example, that much uncertainty exists regarding appropriate thermodynamic data for U and Pu phases.) Only in this way can the model results be interpreted with confidence.

Section 4.3.1 Probability Concepts

4-35 Indicate how correlations between sampled parameters will be identified, quantified, and accounted for in the criticality configuration generation code.

Use of the Monte Carlo method requires that correlations between sampled parameters are taken into consideration if they are not truly independent variables. For example, the drip rate onto the package may affect the WP lifetime. Failure to account for these correlations could result in erroneous results.

Section 4.3.2 Monte Carlo Technique

4-36 Justify the assumption that it is acceptable to only consider the potential for one external criticality for a given realization.

DOE argued that the small probability of a realization yielding a critical configuration obviates the need to analyze the realization for multiple criticalities. This argument is acceptable only if each criticality is an independent event. Since having a single criticality in a realization requires that several sampled parameters are favorable to produce a criticality, additional criticalities are not independent events and the probability of having multiple criticalities for a single realization may not be small enough to be ignored. Failure to consider the potential for multiple criticalities in a realization may lead to an underestimation of the probability of a criticality event occurring.

4-37 Justify the exclusion of water chemical parameters other than pH in regard to item A of the External Criticality list on page 4-39, .

Water chemistry will be greatly altered during Waste package and WF interaction, and concentrations of other components such as carbonate influence the geochemical behavior of U and Pu.

4-38 Clarify how the path selection process does not constitute an additional, non-conservative reduction in probability for a given configuration.

In item B of the External Criticality list, it is stated that random selection of external pathway is weighted according to probability. Subsequent transport modeling utilizes

probability sampling of parameters. It should be made clear that this approach does not constitute redundant application of probability screening of external pathway.

Section 4.3.3 Configuration Generation Code

4-39 Regarding item II.C. of section 4.3.3 on the invert configuration generation code (CGC) geochemistry modeling, justify the exclusion of water chemical parameters other than pH in computing solubility dependence.

In the WP CGC, solubility dependence on other species such as carbonate is included (item I.D.). Such dependence—which, for example, is strong for carbonate content in computing U solubility—should also be included in external cases.

4-40 Clarify how matrix-fracture distribution of water below the WP is calculated (items III.B. and III.C.). Reconcile the distinction between fracture and matrix travel times discussed in section 4.3.3 with the attribution of all flow to the fractures apparent from the discussion in section 4.2.3.

The distinction between matrix and fracture flow has profound implications for modeled travel times and water-rock interaction. For example, it is typically assumed that solutes are not sorbed during fracture flow. The distribution of groundwater flow between the fracture and the matrix will strongly affect U and Pu transport because of contrasting sorption and groundwater travel times. U and Pu transport rates and concentrations are central to models of external criticality.

Section 4.4.1.1 Steady -State Criticality

4-41 Justify the methodology used for analyzing the steady state criticality condition with no iron oxide.

One of the reasons provided in the "Second Waste Package Probabilistic Criticality Analysis: Generation and Evaluation of Internal Criticality Configurations" report for not including no-iron oxide or no-B-10 configurations in the analysis was that the corresponding k_{eff} values are "below any possible range of linearity." This is a questionable basis for excluding these types of realistic configurations. Knowing that taking credit for boron retention in the iron oxide has been dismissed in a later report, it is very possible that the combination of a high acidic environment would cause most of the iron oxide to be dissolved and flushed out of the waste package.

Section 4.4.1.2 Transient Criticality

4-42 Provide an analysis for the seismic event using the time scales such as 0.3 seconds for reactivity insertion as part of the transient criticality analysis.

The cited reference (CRWMS M&O 1997e) does not provide the transient criticality analysis with a duration of 0.3 second as implied in the topical report. The report uses 30 seconds, which is based on the terminal velocity of iron oxide particles, for the duration of reactivity insertion. However, reshuffling of spent fuel in a time duration of

one second or less as the result of the seismic event with no iron oxide must be considered.

4-43 Justify the transient criticality analysis using a computer code which does not have the restrictions that are associated with RELAP5/MOD3.

The one-dimensional RELAP computer code has been developed for reactor cores which have flows parallel to the fuel bundles. The code is not intended to be used for systems with cross flows of more than 10%. First, the validity of using a one dimensional code for two dimensional analysis is not demonstrated. Secondly, the flow in both dimensions in the waste package model are perpendicular to the fuel assemblies for which the RELAP has not been designed. Thirdly, no benchmarks which would demonstrate the degree of applicability and accuracy of RELAP5/MOD3 for the waste package transient criticality conditions, are offered. Other codes which have the capability to perform three dimensional thermal hydraulics analysis might be the more appropriate computer code to use for analyzing the waste package transient conditions.

4-44 Justify the use of computer codes that do not have temperature feedback capability to determine the reactivity of these waste package systems.

The approach proposed in CRWMS M&O 1997e with regard to compensating for the lack of the temperature feedback in MCNP4A, due to unavailability of "an associated cross section library with sufficient temperature data to calculate reactivity changes," does not appear to be very sound. The use of SAS2H, modified by the buckling corrections developed based on MCNP4A, is not accurate. Especially, in deriving the effective radial length of fuel stack, the approach appears to be questionable. The use of another code which has cross section libraries with temperature effects seems to be a more straightforward and accurate way of determining the reactivity insertion as a function of moderator and fuel temperatures.

4-45 Clarify whether in the transient criticality analysis method the code biases and uncertainties, in addition to the Monte Carlo uncertainties, are included in all the k_{eff} values.

Examination of CRWMS M&O 1997b and CRWMS M&O 1997e indicates that the change in reactivity might be based on a subcritical initial condition. This is due to subtracting the code bias and uncertainty from k_{eff} = 1for initial condition. This would result in the majority of reactivity being inserted while configuration is in subcritical condition. Starting with critical condition (i.e., 0.95 + 0.05 for bias and uncertainties) and adding the bias and uncertainties to the other transient conditions (e.g., $k_{eff} = 1.0189 + 2\sigma + 0.05$) would place the reactivity insertion above the critical condition.

4-46 Discuss the approach for transient criticality analysis for high-enriched spent fuels in view of absence of negative Doppler feedback.

The approach in selecting configuration classes for transient criticality presented in the topical report is with respect to low-enriched commercial spent fuel assemblies. The high-enriched spent fuels, such as DOE-owned spent fuels, will have no negative fuel

temperature feedback (i.e., not enough U-238 for Doppler feedback). This type of configuration class must be also analyzed.

4-47 Discuss the overmoderation effect within the waste package in view of the large uncertainty associated with the flow rate into the waste package.

Another transient criticality configuration which must be addressed is the over-moderation configuration. Configurations with large flow rate into the waste package and a subsequent seismic event can result in a positive-feedback criticality.

4-48 Explain how the point neutron kinetics and flow models in RELAP5/MOD3.2 will be adapted for broad applicability to the analysis of internal criticality transients involving all intact and degraded waste forms and packages.

The staff notes that the feedback coefficients in RELAP5/MOD3.2's point neutron kinetics formulation are limited to those feedback mechanisms needed for modeling selected PWR transients. For example, the "void coefficient" formulation assumes that "coolant" and "moderator" are one and the same, which is not valid for certain non-PWR reactor types and likewise not valid for the many intact and degraded waste form/package configurations that involve more than one "moderator/coolant" medium (e.g., see example configurations in Appendix D of the report). The NRC Office of Research, which oversaw the development of RELAP5/MOD3.2 at INEEL, has noted that significant revisions to the code's neutron kinetics and flow models would be needed for applying the code to other reactor types such as CANDU, RBMK (i.e., Chernobyl), etc. Similar revisions may likewise be needed for applying the code to the full range of criticality transients in the repository. This potential code deficiency is closely related to the previously noted deficiencies in the concept of over/under-moderation which does not address the full range of neutronic phenomena that govern positive and negative feedback effects.

4.4.2.1 Steady State Criticality

4-49 Discuss the approach for consequences of external criticality, some of which are presented in *Probabilistic External Criticality Evaluation* report, in the topical report.

The above report presents some qualitative discussion with regard to only an increase in radionuclide inventory. More in-depth quantitative approach is needed to address the steady state external criticality consequence, especially with regard to high enriched spent fuels.

4.4.2.2 Transient Criticality

4-50 Discuss the approach for addressing consequences of transient criticality such as autocatalytic criticality from possible re-concentration of fissile masses in the near field and far field.

Re-concentration of fissile material in the near or far field combined with subsequent sudden flow of water can result in external transient criticality situations. An approach to address possible consequences of this configuration class is needed.

4.4.3.1 Validation of the Steady-State Criticality Consequence Methodology

4-51 Provide a discussion of the approach used to identify applicable experiments which would quantify the bias and uncertainty associated with the steady-state criticality analysis.

Qualitative discussion with respect to conservatism does not provide the quantitative values for uncertainties and bias which need to be identified. For example, examination of the reference material indicates that there is a large uncertainty associated with predicting the steady state power. The analysis indicated that the power produced can be between 0.5 kW and about 4 kW. The average of these two numbers was used. Other areas of analysis have large uncertainties which need to be quantified and taken into consideration for predicting the consequence of the steady-state criticality.

4.4.3.2 Validation of the Transient Criticality Consequence Methodology

4-52 Justify the applicability of RELAP5 to the waste package in light of differences in orientation and presence of iron oxide.

As indicated in the above questions, the RELAP5 has been developed for reactor cores with moderator and coolant flow in the direction parallel to fuel assemblies with minimal cross flow across the fuel assemblies. The situation in the waste package is the reverse of that in the core. Applicable experiments need to be identified in order to provide confidence in predicting transient criticality consequences.

4-53 Provide a discussion of the approach used to identify the super critical experiments which will be used to validate the appropriate transient criticality model.

This section needs to discuss the benchmark experiments which will be used in validating the transient criticality computer code. The discussion should be in terms of area and range of applicability.

APPENDIX A ACRONYMS AND ABBREVIATIONS

No comments

APPENDIX B GLOSSORY

No comments

APPENDIX C EXAMPLE APPLICATION OF THE METHODOLOGY FOR COMMERCIAL SPENT NUCLEAR FUEL

C-1 Provide information on plans for geochemical model validation in this example analysis and the Appendix D example analysis (see discussions above on model validation). This information will make clearer the scope and rigor of the validation approach.

C1.4.3 WASTE FORM DEGRADATION CHARACTERISTICS

C-2 Expand the discussion of the structural and corrosion characteristics of Zircaloy cladding to include the effects of irradiation, reactor water chemistry, operating history, and pre-disposal storage conditions.

Commercial SNF exhibits a wide range of Zircaloy material characteristics, including large variations in the degree of hydriding, oxidation (corrosion), erosion thinning, embrittlement, pellet-cladding interactions, pinhole/crack formation, crud deposition, etc.

C-3 Clarify the intent of the statement that "At sufficiently high temperatures in an oxidizing environment, the fragments will oxidize...."

Oxidation of UO_2 does not, in general, require elevated temperatures.

C3.3 CRITICALITY REGRESSION EXPRESSION

C-4 Justify the applicability of k_∞ which was based on all isotopes in spent fuel, to the spent fuel with the 29 principal isotopes.

The k_∞ regression equation developed by ORNL is based on all the isotopes included in SAS2H. If these results are used for binning the spent fuels with the principal isotope assumption, the k_{eff} appears to be under-predicted.

C-5 Assess the impact of not including axial burnup profile and the reactor operating history bounding parameter values in the regression equations.

Using single uniform axial burnup profile and nominal values for reactor operating history parameters would result in a regression equation under-predicting the k_{eff} values.

C4.1 PROBABILITY ESTIMATION

C-6 Evaluate the potential for axial displacement of the disposal control rods relative to the active fuel.

The topical report takes credit for the control rods as the basis for not considering the most reactive fuel, i.e., fuel with burnup below the loading curve, as part of the population of PWR fuel capable of exceeding the critical limit. Any upset or degradation mechanisms that could produce axial displacement (e.g., tilting of the package or basket) should be identified.

C5.1 CRITICALITY CONSEQUENCE ESTIMATION

C-7 Justify limiting the reactivity insertion scenarios to the relatively slow ones described here.

The reasoning for not considering rapid reactivity insertions resulting from sudden movements such as those caused by collapse of the degrading basket structures, rock falls, etc., should be provided.

C-8 Justify the apparent conclusion that long-term steady-state criticalities bound the consequences of all criticality events.

During a long-term steady-state criticality, many of the radionuclides produced will decay or be burned out. High-power transient events resulting from rapid reactivity insertions and/or autocatalytic effects, perhaps in conjunction with steady-state criticalities, have a potential to produce a burst of short-lived fission products and actinides (i.e., short-half life equates to high activity) as well as transient thermal-mechanical effects that may promote their early release from the repository.

C-9 Verify that short-lived isotopes arising in high-power transients are considered in evaluating dose consequences.

It is not clear that the short-lived isotopes potentially important to the dose consequences of rapid transient events (e.g., Kr-85, I-131, Sr-89, Cs-134) are included in the isotopes evaluated under this methodology. This section limits its evaluation to the 36 TSPA-95 isotopes.

C-10 Justify why the presence of boron and corrosion products dissolved or suspended in the water and that can affect the moderator void coefficient of reactivity is not addressed in the TR.

The presence of absorbers in the "moderator" can produce a positive void reactivity, resulting in autocatalytic feedback effects that are apparently not considered in the proposed methodology.

C-11 Justify the assumption of one-year decay time in assessing the consequences of a fast transient criticality. In particular, explain how the one-year decay time bounds the travel times of all important radionuclides.

Fast transient criticality events resulting from rapid reactivity insertions and/or autocatalytic effects have a potential to produce a burst of short-lived fission products and actinides. However, it is not clear why the one-year decay time assumption is used in assessing the consequences in this case.

C-12 Explain the design modification needed in light of the change in the radionuclide inventory indicated by Table C-16.

The criticality consequence design criterion listed in Section 1.2 states that "the expected radionuclide increase from any criticality event will be less than 10 percent ...." Table C-16 indicates a net increase of more than 18% for the five isotopes which are important to the repository performance. Given the results, verify that a design change is needed.

C6.1 TOTAL SYSTEM PERFORMANCE ASSESSMENT DOSE ESTIMATION

C-13 Evaluate travel times for gases to the surface for the cases of worst-case disruption of the EBS and repository environment from energetic criticality transients.

As stated in the discussion, "gaseous fission products such as 85Kr are ignored because only a small amount are produced. . . . " However, there are others that will be produced in significant amounts. Therefore, an evaluation of the travel times for these gaseous fission products to the surface becomes important.

C-14 Verify that Figures C-34 and C-35 indicate the incremental dose from just increasing the radionuclide inventory by the numbers in Table C-16 in one waste package which is the result of a single criticality. In addition, discuss the risk from multiple waste packages becoming critical because of juvenile failures.

C7.0 CONCLUSION

C-15 The example and the topical report does not address the classes of criticality events with potentially high consequences.

In particular, the report does not give adequate attention to sudden reactivity insertions and the full range of mechanisms for positive reactivity feedback (i.e., autocatalytic criticality).

APPENDIX D EXAMPLE APPLICATION OF THE METHODOLOGY FOR DOE SPENT NUCLEAR FUEL

D2.2 EXTERNAL CONFIGURATIONS

D-1 Correct the reference to the discussion in Appendix C of external configurations. Neither Appendices C or D evaluate external criticality.

D3.1 EVALUATION OF CRITICAL CONFIGURATIONS

D-2 Clarify the application of the methodology in this example, with respect to the flow chart presented in Figure 1.1 in the main body of the report, page 1-10.

It appears, in this example, that design changes were made in the choice of poison material ($GdPO_4$) and its concentration, without having evaluated configurations with respect to the probability criterion. This suggests a departure from the methodology described in the flow chart. The staff notes that the flow chart may need to be revised to

reflect the fact that the need for design changes may become apparent much earlier in the process.

The NRC staff is available to meet and discuss the RAIs at DOE's request. If you have any questions or comments on the above information or wish to schedule a meeting to discuss the RAIs, please contact Sandra Wastler, of my staff, at (301) 415-6724 or by e-mail at SLW1@nrc.gov.

Sincerely,

C. William Reamer, Chief
High-Level Waste and Performance
Assessment Branch
Division of Waste Management
Office of Nuclear Material Safety
and Safeguards

Enclosure: As stated

cc: See attached distribution list

DISTRIBUTION:

File Center	J. Holonich	NMSS r/f	DWM r/f	HLWB r/f
MRahimi	VPerin	CNWRA		

DOCUMENT NAME: S:\DWM\HLWB\SLW\RAI-LTR.WPD
* SEE PREVIOUS CONCURRENCE

OFC	HLWB*		HLWB*		HLWB*		HLWB*		HLWB	
NAME	SWastler: kv		MRahimi		VPerin		NKStablein		CWReamer	
DATE	8/ 12 /99		8/17/99		8/17/99		8/ 16 /99		8/18/99	

OFFICIAL RECORD COPY

ACNW: YES _X_ NO __ Delete file after distribution: Yes_X_ No __

1) This document should be made available to the PUBLIC __SLW__ 8 /12 /99
(Initials) (Date)

2) This document is/is not related to the HLW program. If it is related to HLW, it should be placed in the LSS. _SLW__ 8 /12 /99

The NRC staff is available to meet and discuss the RAIs at DOE's request. If you have any questions or comments on the above information or wish to schedule a meeting to discuss the RAIs, please contact Sandra Wastler, of my staff, at (301) 415-6724 or by e-mail at SLW1@nrc.gov.

Sincerely,

C. William Reamer, Chief
High-Level Waste and Performance
Assessment Branch
Division of Waste Management
Office of Nuclear Material Safety
and Safeguards

Enclosure: As stated

cc: See attached distribution list

DISTRIBUTION:

File Center	J. Holonich	NMSS r/f	DWM r/f	HLWB r/f
MRahimi	VPerin	CNWRA		

DOCUMENT NAME: S:\DWM\HLWB\SLW\RAI-LTR.WPD
* SEE PREVIOUS CONCURRENCE

OFC	HLWB*		HLWB		HLWB		HLWB*		HLWB	
NAME	SWastler: kv		MRahimi		VPerin Vanice Perin		NKStablein		CWReamer	
DATE	8/ 12 /99		8/17/99		8/17 /99		8/ 16 /99		8/ /99	

OFFICIAL RECORD COPY

ACNW: YES _X_ NO __ Delete file after distribution: Yes X No __

1) This document should be made available to the PUBLIC __SLW__ 8 /12 /99
(Initials) (Date)

2) This document is/is not related to the HLW program. If it is related to HLW, it should be placed in the LSS. _SLW__ 8 /12 /99

The NRC staff is available to meet and discuss the RAIs at DOE's request. If you have any questions or comments on the above information or wish to schedule a meeting to discuss the RAIs, please contact Sandra Wastler, of my staff, at (301) 415-6724 or by e-mail at SLW1@nrc.gov.

Sincerely,

C. William Reamer, Chief
High-Level Waste and Performance
Assessment Branch
Division of Waste Management
Office of Nuclear Material Safety
and Safeguards

Enclosure: As stated

cc: See attached distribution list

DISTRIBUTION:

File Center	J. Holonich	NMSS r/f	DWM r/f	HLWB r/f
MRahimi	VPerin	CNWRA		

DOCUMENT NAME: S:\DWM\HLWB\SLW\RAI-LTR.WPD

OFC	HLWB		HLWB		HLWB				
NAME	SWastler:kv		NKStablein		CWReamer				
DATE	8/12/99		8/16/99		8/ /99				

OFFICIAL RECORD COPY

ACNW: YES ✓ NO ___ Delete file after distribution: Yes ✓ No ___

1) This document should/should not be made available to the PUBLIC SLW 8/12/99
(Initials) (Date)

2) This document is/is not related to the HLW program. If it is related to HLW, it should/should not be placed in the LSS. SLW 8/12/99
(Initials) (Date)