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June 26, 2000

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SUBJECT: SAFETY EVALUATION REPORT FOR DISPOSAL CRITICALITY ANALYSIS METHODOLOGY TOPICAL REPORT, REVISION 0

Dear Mr. Brocoum:

By your letter, dated January 7, 1999, the U.S. Department of Energy (DOE) requested that the U.S. Nuclear Regulatory Commission (NRC) review the Disposal Criticality Analysis Methodology Topical Report, Revision 0 (YMRP/TR-004Q). The NRC staff has completed its review and has prepared the enclosed safety evaluation report (SER) documenting the results of the staff evaluation of the Topical Report.

The scope of this SER is based on the revised scope of the Topical Report provided in your letter, dated November 19, 1999, transmitting the DOE's responses to the NRC's request for additional information. In issuing this SER, the NRC staff also considered the DOE's oral comments made in the March 22, 2000, technical exchange, and the written comments, transmitted to the NRC by DOE's letter, dated May 17, 2000, on the draft SER .

This SER documents the results of staff evaluation in terms of Acceptance, Confirmatory Items, or Open Items. The review and acceptance of the DOE's proposed approach is specifically within the context and scope of the request. The Confirmatory Items pertain to those specific areas for which the NRC finds the DOE's approach or methodology acceptable provided that the DOE fulfills its commitments to take into account additional factors identified by the NRC staff. These factors are identified as conditions for acceptance. The Open Items are related to those specific areas in which the staff does not agree with the DOE's approach or where there is not sufficient information for the staff to make a determination.

As stated in your November 19, 1999 letter, "The DOE plans to seek acceptance of additional aspects of the methodology through a revision to the Topical Report planned for submittal late in Fiscal Year 2000." The staff hopes to close most of the open items after examining the revised Topical Report.

If you have any questions regarding this letter, please contact Meraj Rahimi at (301) 415-6616.

Sincerely,

C. William Reamer, Chief
High-Level Waste Branch
Division of Waste Management
Office of Nuclear Material Safety
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Enclosure: Safety Evaluation Report for Disposal Criticality Analysis Methodology Topical Report, Revision 0

cc: See attached list

Letter to S.J. Brocoum from C.W. Reamer dated: June 26, 2000

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# Safety Evaluation Report for Disposal Criticality Analysis Methodology Topical Report, Revision 0

June 2000

U.S. Nuclear Regulatory Commission
Office of Nuclear Material Safety and Safeguards
Washington, DC 20555-0001

## ABSTRACT

In January 1999, the U.S. Department of Energy (DOE)/Office of Civilian Radioactive Waste Management (OCRWM) submitted the Disposal Criticality Analysis Methodology Topical Report, Revision 0 (TR) to the U.S. Nuclear Regulatory Commission (NRC) for review and approval. The TR presents an overall approach for consideration of post-closure disposal criticality of commercial and defense high-level waste to be placed at the proposed Yucca Mountain site. During the course of the review and interactions with the DOE, through a series of technical exchanges and a request for additional information (RAI), the scope of the TR was focused on specific aspects of the methodology. This safety evaluation report (SER) presents the results of the NRC staff evaluation of those specific aspects of the methodology. The NRC staff has presented the results of its evaluation in terms of Acceptance, Confirmatory Items, or Open Items throughout the SER. The Open Items from all the sections are listed in the Conclusion section.

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## ABBREVIATIONS

| | |
|---|---|
| AC | acceptance criteria |
| ANS | American Nuclear Society |
| ANSI | American National Standard Institute |
| AOA | area of applicability |
| BOL | beginning-of-life |
| BWR | boiling-water reactor |
| CL | Critical Limit |
| CLST | container life and source term |
| CRC | commercial reactor criticals |
| CRWM | Civilian Radioactive Waste Management |
| DFTL | Distribution Free Tolerance Limit |
| DOE | U.S. Department of Energy |
| EBS | engineered-barrier system |
| ENDF | Evaluated Nuclear Data File |
| ENFE | evolution of the near-fields |
| EOC | end-of-cycle |
| EPA | U.S. Environmental Protection Agency |
| FEPS | features, events, processes |
| FF | far field |
| FM | fissile material |
| HLW | high-level waste |
| IRSR | issue resolution status report |
| ISI | integrated subissue |
| KTI | key technical issue |
| LA | license application |
| LANL | Los Alamos National Laboratory |
| LCE | laboratory critical experiments |
| LLNL | Lawrence Livermore National Laboratory |
| LUTB | Lower Uniform Tolerance Band |
| MOC | middle-of-cycle |
| MOX | mixed oxide |
| mrem | millirem |
| MSL | master scenario list |
| mSv | milliSievert |
| NDTL | Normal Distribution Tolerance Limit |
| NF | near field |
| NNPP | Naval Nuclear Propulsion Program |
| NRC | U.S. Nuclear Regulatory Commission |
| PA | performance assessment |
| PRA | probabilistic risk assessment |
| PWR | pressurized-water reactor |
| QA | quality assurance |

| | |
|---|---|
| RAI | request for additional information |
| RM | review method |
| ROA | range of applicability |
| RT | radionuclide transport |
| SER | safety evaluation report |
| SNF | spent nuclear fuel |
| THC | thermo-hydraulic coupling |
| TR | topical report |
| TSPA | total system performance assessment |
| VA | Viability Assessment |
| WF | waste form |
| WP | waste package |
| YM | Yucca Mountain |
| YMRP | Yucca Mountain Review Plan |

# 1 INTRODUCTION

In January 1999 the DOE submitted to the NRC, the "Disposal Criticality Analysis Methodology Topical Report" (TR), YMP/TR-004Q Rev. 0 (U.S. DOE, YMPR/TR-004Q, Rev. 0). After a preliminary review by NRC staff, the TR was formally accepted for a detailed technical review (Letter, February 11, 1999). In August 1999, a RAI was issued to the DOE by the NRC (Letter, August 18, 1999). The response to the RAI was submitted to the NRC by the DOE in November 1999 (DOE letter dated November 19, 1999) and was used in the preparation of this SER.

## 1.1 Background

The DOE is currently characterizing the Yucca Mountain (YM), Nevada, site to determine if the site can be used for deep underground emplacement of spent nuclear fuel (SNF) from commercial nuclear power plants; DOE-owned SNF; DOE high-level waste (HLW) glass canisters; and Naval SNF discharged from Navy ships. If the DOE chooses YM site for emplacement of SNF and HLW, the DOE must obtain a license from the NRC by complying with the requirements in the proposed 10 CFR Part 63 (NRC, 1999a) which is the implementation regulation developed based on the U. S. Environmental Protection Agency's (EPA's) proposed 40 CFR Part 197 (EPA, 1999).

As part of the pre-licensing activities, the DOE and the NRC are currently interacting to determine what the important licensing issues are and how they can be resolved. One of the ways of interaction between the two agencies is TR submittal by the DOE and subsequent evaluation by the NRC. The TR Review Plan (Letter, February 28, 1994) describes the procedure and the criteria for addressing important-to-safety subjects in the form of TRs before the license application (LA).

One such subject is post-closure disposal criticality. In the TR, the DOE has presented a methodology for minimizing the potential for, and consequences of, criticality, which would determine the design of the criticality control system of the engineered barrier system (EBS). The DOE is planning to use this methodology to demonstrate the potential for, and the consequence of, post-closure criticality in the license application.

## 1.2 Purpose

The DOE has stated that the "...fundamental objective of this topical report is to present the planned risk-informed disposal criticality analysis methodology to the NRC to seek acceptance that the principles of the methodology and the planned approach to validating the methodology are sound." Therefore, the purpose of this SER is to document the staff evaluation of the DOE's proposed risk-informed disposal criticality analysis methodology and to document the staff's conclusion on using the methodology

to demonstrate the impact of post-closure criticality on the YM site in the license application.

For the license application phase, the DOE may use those parts of the methodology that are approved in this SER toward determination of the potential for post-closure criticality and the consequences of the credible criticality events at YM. As part of the LA process, the DOE will demonstrate that the methodology is valid for specific waste forms, waste package designs, and repository conditions.

The staff's review of the TR, within the scope described in Section 1.3, and the staff's conclusions regarding the acceptability of the disposal criticality approach to be used by the DOE, are documented in this SER.

## 1.3 Scope

Section 1.2 of the TR lists the DOE's specific approval requests. However, as the result of two telephone conferences (Rahimi, personal communication; Wastler, personal communication) and two technical exchanges (Letter, September 2, 1999; Letter, October 28, 1999), the DOE's requests were modified, clarified, and specified further. These changes to the requested items were submitted to the NRC as part of the responses to the NRC's RAI. The changes to the requested items are summarized and labeled as "near-term acceptance" in Enclosure 2 of the responses to the RAI. Therefore, the scope of this evaluation is within the bounds requested by the DOE, in Enclosure 2 of the responses to the RAI dated November 19, 1999, with references to Section 1.2 of the TR.

Generally, the staff evaluation was limited to assessing the proposed overall methodology and the approach in modeling and validation. Any requests related to specific application, data, computer codes, specific benchmark experiments, specific isotope, or any examples were considered to be beyond the scope of the evaluation for the purposes of this SER. Those requests pertaining to: (1) the areas in which the staff does not agree with the DOE's approach; or (2) where there is insufficient information for the staff to make a determination; or (3) areas that are not part of the DOE's near-term acceptance are listed in the pertinent sections and in the "Conclusions" section as Open Items. Other aspects of the methodology, such as specific models and their validation, will be part of the LA, therefore, they were not included in the TR and are not considered to be open items.

The staff evaluated the methodology with respect to its application to commercial Pressurized Water Reactor (PWR), Boiling Water Reactor (BWR), and Mixed Oxide (MOX) SNF, and to some extent for a range of waste conditions (intact, degraded, and degradation product) and different possible locations (in-package, near-field, and far-field) as requested by the DOE. The proposed methodology in the TR is mainly centered around commercial SNF, especially with respect to the validation approach for

neutronics models. Therefore, the staff will evaluate the applicability of any parts of the proposed methodology to DOE-owned SNF, immobilized plutonium, and vitrified HLW glass, when the DOE submits addendums to TR for these waste types. For Naval SNF in an intact condition, the staff is currently evaluating a methodology that the Naval Nuclear Propulsion Program (NNPP) has described in a separate submittal as an addendum to the TR (Department of Navy, October 1999).

# 2 STAFF EVALUATION CRITERIA

To evaluate DOE's proposed approach in analyzing the potential for and consequences of criticality events at the proposed YM repository, a set of documents has been developed that establishes the requirements that must be met to demonstrate compliance with the applicable regulations. The following sections provide a brief description of these pertinent documents. In addition, the regulatory requirements and guidance related to disposal criticality contained in each of these documents are presented.

## 2.1 Proposed U.S. Code of Federal Regulations, Title 10, Part 63

The overall licensing criteria for the disposal of SNF and HLW in the proposed geologic repository at YM, Nevada, are included in the proposed U.S. Code of Federal Regulations (CFR), Title 10, Part 63 (Nuclear Regulatory Commission, 1999a). This proposed regulation was published for public comment before the February 22, 1999, issue of the Federal Register. The NRC staff have addressed the public comments and the Commission is currently reviewing the proposed final rule. Furthermore, the EPA has issued the proposed 40 CFR Part 197 (U.S. Environmental Protection Agency, 1999) for public comment. The proposed Part 197 establishes the public health and safety standard for radioactive material stored or disposed of in the potential repository at YM, Nevada. Under the Energy Policy Act of 1982, NRC regulations in Part 63 must be consistent with the final EPA Standard for YM. Depending on the final resolution of the public comments on both the proposed Part 63 and the proposed Part 197, some of the quantitative measures provided in the proposed Part 63 may change either in the final rule or as amendments to the final rule.

The proposed Part 63 specifies the overall performance objectives for the preclosure and postclosure phases of the repository. The DOE TR, however, only addresses postclosure criticality. The specification for overall performance of the repository with respect to postclosure is in terms of expected annual dose to the average member of the critical group. In keeping with the Commission philosophy of risk-informed, performance-based regulation, there are no specific design criteria for postclosure criticality control in the proposed 10 CFR Part 63. Criticality is one of the processes or events that must be considered in the assessment of the overall system performance. The quantitative aspect of the requirement for the overall system performance assessment is stated in 10 CFR 63.113.

> *63.113 (b) The engineered barrier system shall be designed so that, working in combination with natural barriers, the expected annual dose to the average member of the critical group shall not exceed 0.25 mSV (25 mrem) TEDE at any time during the first 10,000 years after permanent closure, as a result of radioactive materials released from the geologic repository.*

The quantitative aspect of the overall system performance with respect to probability is prescribed in 10 CFR 63.114.

> *63.114 Any performance assessment used to demonstrate compliance with §63.113 shall: . . . (d) Consider only events that have at least one chance in 10,000 of occurring over 10,000 years. (e) Provide the technical basis for either inclusion or exclusion of specific features, events, and processes of the geologic setting in the performance assessment. Specific features, events, and processes of the geologic setting must be evaluated in detail if the magnitude and time of the resulting expected annual dose would be significantly changed by their omission. (f) Provide the technical basis of either inclusion or exclusion of degradation, deterioration, or alteration processes of engineered barriers in the performance assessment, including those processes that would adversely affect the performance of natural barriers. Degradation, deterioration, or alteration processes of engineered barriers must be evaluated in detail if the magnitude and time of the resulting expected annual dose would be significantly changed by their omission.*

The staff has used the foregoing requirements in evaluating the DOE proposed approach for addressing the postclosure criticality aspect of SNF and HLW disposal in the YM. These requirements are translated into specific acceptance criteria (AC) and review methods (RMs) for disposal criticality control subsystems described in the various issue resolution status reports (IRSRs) (U.S. Nuclear Regulatory Commission, 1999b–d, 2000). Eventually all the AC and RMs for criticality control subsystems will be included in the YM Review Plan (YMRP). The IRSRs will contain only the status of the issue resolution and the technical basis.

## 2.2 Draft YMRP Plan

The draft YMRP is currently being developed at the NRC. This document will be used to review the DOE's LA to ensure that the LA meets the requirements of Part 63. The YMRP will be used to assess the adequacy of both the preclosure and postclosure safety assessments performed for the repository system. The YMRP ensures that the LA demonstrates the site will meet all the requirements of Part 63.

Review of the postclosure performance assessment using the YMRP will be based on an integrated subissue (ISI) framework. The ISIs are the integrated features, events, and processes (FEPs) that could impact system performance. The effects of postclosure criticality will be addressed under the following ISIs in the YMRP: "Degradation of Engineered Barriers"; "Radionuclide Release Rates and Solubility Limits"; "Radionuclide Transport in the Unsaturated Zone, Radionuclide Transport in the Saturated Zone"; and "Mechanical Disruption of Engineered Barriers." Other aspects of postclosure criticality, such as identification of criticality FEPS with probabilities > $10^{-8}$ will be addressed in other parts of the YMRP.

The potential for criticality during preclosure operations will be assessed in the preclosure section of the YMRP. The proposed methodology in the DOE TR, however, applies only to assessing the probability and consequences of a postclosure criticality event and does not apply to preclosure criticality. Therefore, preclosure criticality will not be discussed in this SER.

## 2.3 IRSRs

The NRC strategic planning assumptions call for the early identification and resolution, at the staff level, of issues before the receipt of a potential LA to construct a geologic repository. The principal means for achieving this goal is through informal, prelicensing consultations with the DOE. These consultations, required by law, occur in an open manner that permits observation by the State of Nevada, Tribal Nations, affected units of local government, and interested members of the public. Obtaining input and striving for consensus from the technical community and interested parties help the issue resolution process. The issue resolution approach reduces the number of, and better defines, issues that may be in dispute during the NRC licensing review.

Thus, consistent with the NRC regulations and a 1993 agreement with the DOE, staff-level issue resolution can be achieved during the prelicensing consultation period; however, such resolution at the staff level would not preclude the issue being raised and considered during licensing proceedings. Issue resolution at the staff level during prelicensing is achieved when the staff has no further questions or comments concerning the DOE approach to addressing an issue. There may be some cases where resolution at the staff level may be limited to documenting a common understanding regarding differences in the NRC and DOE technical positions. Pertinent, additional information could raise new questions or comments regarding a previously resolved issue.

An important step in the staff's approach is to provide the DOE with feedback regarding issue resolution before the forthcoming Site Recommendation and LA. IRSRs are the primary mechanism that the NRC staff will use to provide the DOE with feedback on key technical issues (KTIs). IRSRs focus on: (i) AC for issue resolution; and (ii) the status of resolution, including areas of agreement or where the staff has comments or questions. Open meetings and technical exchanges with the DOE have provided, and will continue to provide, additional opportunities to discuss issue resolution, identify areas of agreement and disagreement, and develop plans to resolve such disagreements. An important goal of these prelicensing interactions is to exchange sufficient information to reach closure on the open items by the time the DOE submits the LA.

### 2.3.1 Total System Performance Assessment and Integration

The DOE plans to treat postclosure criticality as a disruptive scenario in the performance assessment conducted for YM. The treatment of criticality within the total system performance assessment (TSPA) is considered acceptable if the following AC from the total system performance assessment and integration (TSPAI) IRSR (Nuclear Regulatory Commission, 2000) are met:

1. *DOE has identified a comprehensive list: (i) of processes and events that are present or might occur in the YM region, and (ii) that includes those processes and events that have the potential to influence repository performance.*

2. *DOE has provided adequate documentation identifying how its initial list of processes and events have been grouped into categories.*

3. *Categorization of processes and events is compatible with the use of categories during the screening of processes and events.*

4. *The probability assigned to each category of processes and events is consistent with site information, well-documented, and adequately considers uncertainty.*

5. *Processes and events screened from the PA, on the basis of their probability of occurrence, have been demonstrated to have a probability of less than one chance in 10,000 of occurring over 10,000 years.*

6. *DOE has demonstrated that categories of processes and events omitted from the PA based on low consequence would not significantly change the calculated expected annual dose.*

7. *DOE has provided adequate documentation identifying: (i) whether processes and events have been addressed through consequence model abstraction or scenario analysis, and (ii) how the remaining categories of processes and events have been combined into scenario classes.*

8. *The set of scenario classes identified by DOE is mutually exclusive and complete.*

9. *Scenario classes that are not credible for the YM repository because of waste characteristics, repository design, or site*

*characteristics—individually or in combination—have been identified and sufficient justification has been provided for DOE's conclusions.*

10. *The probability assigned to each scenario class is consistent with site information, well-documented, and appropriately considers uncertainty.*

11. *DOE has demonstrated that for scenario classes screened from the PA on the basis of their probability of occurrence: (i) the probability used for screening the scenario class is defined from combinations of initiating processes and events, and (ii) they have a probability of less than one chance in 10,000 of occurring over 10,000 years.*

12. *Scenario classes omitted from the PA on the basis of low consequence have been demonstrated to not significantly change the calculated expected annual dose.*

If criticality is included in the TSPA analyses, the model abstraction for criticality should meet the five AC in the YMRP on model abstraction:

13. *Sufficient data (e.g., field, laboratory, and natural analog data) are available to adequately define relevant parameters and conceptual models necessary for developing the abstraction in the TSPA. Where adequate data do not exist, other information sources such as expert elicitation have been appropriately incorporated into the TSPA.*

14. *Parameter values, assumed ranges, probability distributions, and bounding assumptions used in the TSPA abstraction are technically defensible and reasonably account for uncertainties and variabilities.*

15. *Alternative modeling approaches consistent with available data and current scientific understanding are investigated and results and limitations are appropriately considered in the abstractions.*

16. *Models implemented in the TSPA provide results consistent with output of detailed process-level models or empirical observations (e.g., laboratory testing, field measurements, or natural analogs).*

17. *TSPA adequately incorporates important design features, physical phenomena and couplings, and uses consistent and appropriate assumptions throughout the abstraction process.*

Additionally, all data and computer codes used in the criticality abstraction should meet the following acceptance criterion related to quality assurance (QA):

18. *"The collection, documentation, and development of data, models, and computer codes have been performed under acceptable quality assurance (QA) procedures, or if the data, models, or computer codes were not subject to an acceptable QA procedure, they have been appropriately qualified."*

Finally, formal expert elicitation can be used to support data synthesis and model development for DOE's criticality abstraction, provided that the elicitation meets the following acceptance criterion, or the DOE can demonstrate, in other ways, that the conduct of the elicitation provides an adequate level of confidence in the results:

19. *"Formal expert elicitation used to support data synthesis and model development for DOE's TSPA have been conducted and documented under acceptable procedures."*

### 2.3.2 Container Life and Source Term

The container life and source term (CLST) IRSR (Nuclear Regulatory Commission, 1999b) contains AC related to how the conditions inside the waste package (WP) could influence the occurrence of criticality and how in-package criticality could affect WP and engineered barrier subsystem performance. The evaluation of the probability and consequences of in-package criticality on WP and engineered barrier subsystem performance will be acceptable if the following AC are met:

1. *Mathematical model limitations and uncertainties in modeling were defined and documented.*

2. *Primary and alternative modeling approaches consistent with available data and current scientific understanding were investigated and their results and limitations considered in evaluating the subissue.*

3. *DOE has used sound technical bases for selecting the design criteria for components to mitigate any potential effects of in-package criticality on the repository performance. These design criteria may include development of subcritical limit, probability and consequence of criticality, and any other design criteria considered to be necessary by DOE.*

4. *DOE has identified all the features, events, and processes that may increase the reactivity of the system inside the WP.*

5. *DOE has identified the configuration classes and configurations that have potential for nuclear criticality. If models are used to develop the configuration, approach and accuracy in modeling verification and validation will be evaluated.*

6. *DOE has developed a technically defensible, transparent, and traceable method in assigning probability values to each of the scenario classes, scenarios, configuration classes, and configurations.*

7. *DOE has developed appropriate computer models, input parameters, and determined quantitative values for calculating the effective neutron multiplication factor ($k_{eff}$), including appropriate biases and uncertainties in the model.*

8. *DOE has developed appropriate computer models, evaluated input parameters, and determined quantitative values for calculating the radionuclide inventory, heat, kinetic energy, and other parameters that would change as a result of $k_{eff}$ exceeding the subcritical limit developed under Criterion (3).*

9. *DOE has determined the risk contribution from the in-package criticality to the total repository system performance appropriately.*

### 2.3.3 Evolution of the Near-Field Environment

The evolution of the near-field (NF) environment IRSR (Nuclear Regulatory Commission, 1999c) contains AC related to how the NF conditions could influence the occurrence of criticality and how nuclear criticality outside of the WP affects the NF environment. The evaluation of the probability and consequences of criticality in the NF environment will be acceptable if the following AC are met:

1. *Sensitivity and uncertainty analyses (including consideration of alternative conceptual models) were completed to determine whether criticality will impact repository performance, and whether additional new data are needed to better define ranges of input parameters.*

2. *Available data relevant to both temporal and spatial variations in conditions affecting coupled (THC) effects on the potential for nuclear criticality in the near-field environment were considered.*

3. *DOE's evaluation of coupled THC processes properly considered site characteristics in establishing initial and boundary conditions for conceptual models and simulations of coupled processes that may affect nuclear criticality in the near-field environment.*

4. *Sufficient data were collected on the characteristics of the natural system and engineered materials, such as the type, quantity, and reactivity of material, in establishing initial and boundary conditions for conceptual models and simulations of THC coupled processes that may affect nuclear criticality in the near-field environment.*

5. *Reasonable or conservative ranges of parameters or functional relations were used to determine effects of coupled THC processes on potential nuclear criticality in the near-field environment. Parameter values, assumed ranges, probability distribution, and bounding assumptions are technically defensible and reasonably account for uncertainties.*

6. *Uncertainty in data due to both temporal and spatial variations in conditions affecting coupled THC effects on potential nuclear criticality was considered.*

7. *DOE's evaluation of coupled THC processes properly considered the uncertainties in the characteristics of the natural system and engineered materials, such as the type, quantity, and reactivity of material, in establishing initial and boundary conditions for conceptual models and simulations of THC coupled processes that affect potential nuclear criticality.*

8. *The initial conditions, boundary conditions, and computational domain used in sensitivity analyses involving coupled THC effects on potential nuclear criticality in the near-field environment were consistent with available data.*

9. *Alternative modeling approaches consistent with available data and current scientific understanding were investigated, and their results and limitations were appropriately considered.*

10. *DOE provided a reasonable description of the mathematical models included in its analyses of coupled THC effects on potential nuclear criticality. The description should include a discussion of alternative modeling approaches not considered in its final analyses and the limitations and uncertainties of the chosen model.*

11. *The mathematical models for coupled THC effects on potential nuclear criticality are consistent with conceptual models based on inferences about the near-field environment, field data and natural alteration observed at the site, and expected engineered materials.*

12. *DOE appropriately adopted, accepted, and well-documented, procedures to construct and test the numerical models used to simulate coupled THC effects on potential nuclear criticality.*

13. *Abstracted models for coupled THC effects on potential nuclear criticality were based on the same assumptions and approximations shown to be appropriate for closely analogous natural or experimental systems. Abstracted model results were verified through comparison to outputs of detailed process models and empirical observations. Abstracted model results are compared with different mathematical models to judge robustness of results.*

14. *DOE has considered all the relevant features, events, and processes. The abstracted models adequately incorporated important design features, including criticality safety features; physical phenomena and couplings, including neutron absorbers; and used consistent and appropriate assumptions throughout.*

15. *Important mass transfer and mass transport processes and mechanisms considered for formation of both a critical mass and configuration are plausible for the YM near-field environment.*

16. *Models reasonably accounted for known temporal and spatial variations in conditions affecting coupled THC effects on potential nuclear criticality.*

17. *Criticality in the near field, and not all THC couplings, may be determined to be important to performance, and DOE may adopt assumptions to simplify PA analyses. If potentially important couplings and criticality in the near field are neglected, DOE should provide a technical basis for doing so. The technical basis could include activities, such as independent modeling, laboratory or field data, or sensitivity studies.*

18. *Where simplifications for modeling coupled THC effects on potential nuclear criticality were used for PA analyses instead of detailed process models, the bases used for modeling assumptions and approximations were documented and justified.*

19. *Data and models were collected, developed, and documented under acceptable QA procedures.*

20. *Deficiency reports concerning data quality on issues related to coupled THC effects on the potential for nuclear criticality were closed.*

21. *If used, expert elicitations were conducted and documented in accordance with the guidance in NUREG-1563 (Nuclear Regulatory Commission, 1996) or other acceptable approaches.*

### 2.3.4 Radionuclide Transport

The radionuclide transport IRSR (Nuclear Regulatory Commission, 1999d) contains AC related to how far-field (FF) conditions could influence the occurrence of criticality and how nuclear criticality outside the WP could affect the FF environment. The evaluation of the impacts of criticality outside of the repository drifts will be acceptable if the following ACs are met:

1. *The DOE has determined the probabilities of scenarios that lead to the accumulation of a critical mass of fissile material into a critical configuration within 10,000 years in the farfield using appropriate site characteristics.*

2. *For those scenarios that have probabilities greater than $10^{-7}$, the DOE has determined their effect on performance.*

## 2.4 Applicable Regulatory Guides and Standards

There are no formal regulatory guidance documents or industry standards specific to criticality in a permanent HLW repository. Existing guides may be followed in formulating and assessing the disposal criticality analyses. In conducting the review of the TR, the staff has referred to Regulatory Guide 3.71, "Nuclear Criticality Safety Standards for Fuels and Material Facilities" (U.S. Nuclear Regulatory Commission, 1998). For this review, staff followed this guide only with respect to issues of performance; staff may follow aspects related to design in conducting pre-closure reviews. Regulatory Guide 3.71 is appropriate because it concerns criticality analyses supporting safety in handling, storage, and transportation outside reactors. Regulatory Guide 3.71 replaces several earlier guides (3.1; 3.4; 3.43; 3.45; 3.47; 3.57; 3.58; 3.68; 3.70; and 8.12) and recommends that licensees follow procedures outlined in several American National Standards Institute (ANSI)/American Nuclear Society (ANS)–8 nuclear criticality safety standards, supplemented by a detailed, operation-specific criticality analysis.

In developing the criticality methodology, the DOE has applied guidance from NRC regulatory guides as well as from industry standards. From NUREG/CR–2300 (U.S. Nuclear Regulatory Commission, 1983), the DOE has borrowed the approach to

probabilistic risk assessment (PRA) for nuclear power plants and modified it for postclosure disposal criticality analysis. The three steps of PRA—scenario analysis, failure and release, and environmental transport and exposure—may be mapped in general terms to the three key steps of the disposal methodology: scenario analysis, source term, and input of source term into TSPA. Two other NUREGs—NUREG/CR–6361 (Lichtenwalter, et al., 1997) and NUREG/CR–5661 (U.S. Nuclear Regulatory Commission, 1997)—have been used for criticality model benchmarking and establishing criticality limits, respectively. These two NUREGs cite a subset of the ANSI/ANS standards discussed by the TR: ANSI/ANS–8.1, 8.15, 8.17, and 8.10, which are concerned with criticality control of fissionable materials outside of reactors (e.g., during transport or in nonreactor facilities). Application of the standards to disposal criticality is not direct, but is established by analogy. Finally, the TR cites two NRC Regulatory Guides: 3.4 (U.S. Nuclear Regulatory Commission, 1986a) and 3.58 (U.S. Nuclear Regulatory Commission, 1986b), which are again concerned with criticality analyses supporting safety in handling, storage, and transportation. These regulatory guides endorse ANSI/ANS–8.1 and -8.17. The TR states that the disposal criticality methodology is consistent with these guides with the exception of the TR approach to burnup confirmation. The DOE states it does not plan to necessarily measure reactivity on each fuel bundle and may rely on burnup inferences (see last paragraph of this section).

In the absence of formal regulatory guidance documents or industry standards applicable to permanent HLW disposal, the DOE supports application of these materials by analogy. For example, a level 1 PRA "... consists of an analysis of plant design and operation focused on the accident sequences that could lead to core melt, their basic causes, and their frequencies" (U.S. Nuclear Regulatory Commission, 1983). This analysis is deemed analogous to the DOE method for constructing scenarios and configuration classes that define the possible ways in which criticality may occur in the repository. The DOE cites cases in which strict adherence to the standards is inappropriate. For example, in applying ANSI/ANS–8.1, single-parameter limits and the double-contingency criterion are deemed inadequate for a complex repository system. The DOE does not necessarily intend to strictly follow regulatory guidance and industry standards because they are not specific to a repository system. Justification for model approaches, parameters, limits, and validations are described elsewhere in the TR, with the cited guidance and standards providing starting points for development of the DOE approach.

Recently, NRC Regulatory Guides 3.4 and 3.58 have been subsumed into Regulatory Guide 3.71, with no changes in licensing commitments or requirements. In its response to NRC RAI 2-1, the DOE stated that it will revise Section 2.3.3 of the TR to refer to the newer Regulatory Guide 3.71 in place of other regulatory guides.

The NRC approves of the DOE approach to the use of regulatory guides and standards applicable to performance in the criticality analysis, with one condition. NRC Regulatory Guide 3.71 (U.S. Nuclear Regulatory Commission, 1998) states, in Section C, that

safety criteria for preventing nuclear criticality described in ANSI/ANS–8.17–1997 are acceptable as guidance, with one exception. This exception is that "credit for fuel burnup may be taken only when the amount of burnup is confirmed by physical measurements that are appropriate for each type of fuel assembly in the environment in which it is to be stored." In contrast, the DOE states, in Section 2.3.3 of the TR, that "... it has not been determined whether such measurements are needed for all spent fuel bundles, for suitable samples, or are not needed." This departure from Regulatory Guide 3.71 is not acceptable, and this issue is considered an open item.

## Open Item

- The staff believes that burnups of spent fuel assemblies must be verified through measurements before their loading into the waste packages for the purpose of burnup credit verification.

# 3 STAFF EVALUATION

## 3.1 Overall Methodology (Section 1.5)

Section 1.5 of the TR provides a description of the methodology that the DOE will use to assess the probability and consequences of criticality in the Yucca Mountain repository during the postclosure period. In its responses to the RAI, DOE has requested an approval of the overall methodology as stated by the DOE in the following:

> *DOE requests acceptance of the risk-informed processes that is the core of the methodology. The risk-informed process is illustrated in Figure 1-1 (discussed in Section 1.5) and revised in Attachment B of Enclosure 1. We do not seek acceptance for a specific application of the methodology, and we understand that we will need to demonstrate acceptability of specific applications to support licensing.*

Attachment B of Enclosure 1 of the responses to the RAI depicts the overall methodology. This flow chart is shown in Figure 3.1-1 of this SER. In its approach, the DOE uses the waste form characteristics; WP/EBS; the site characteristics; the WP degradation characteristics; and the Master Scenario List as input to the WP/EBS degradation process. Once the configurations are determined through the degradation processes, the criticality analyses are performed for these configurations. Using four decision points, shown in the diamond-shaped boxes, which correspond to the four design criteria, the DOE screens configurations for any further evaluation, or performs design modifications. The four decision points are the four design criteria for which the DOE has requested approval and which the NRC staff has evaluated in the following sections.

For those configurations that may result in critical conditions, the DOE either modifies the design or performs consequence analysis, depending on the probability. For those critical configurations whose probabilities are lower than the Probability Criterion, the DOE proposes to perform consequence analysis, and for the ones higher than the Probability Criteria, design modifications are performed to reduce $k_{eff}$. The method calls for performing a TSPA estimation of dose increment for those critical configurations whose consequences exceed the Criticality Consequence Criteria. If the incremental dose does not cause the total dose to exceed the Repository Performance Objective, the WP design becomes acceptable with respect to criticality; otherwise, the criticality control system of the WP is redesigned.

The staff evaluation of the DOE's request with respect to the overall methodology, which uses a risk-informed approach, was performed using the proposed Part 63. As mentioned earlier, the proposed Part 63 is subject to change pending the resolution of public comments and the finalization of the EPA-proposed Part 197.

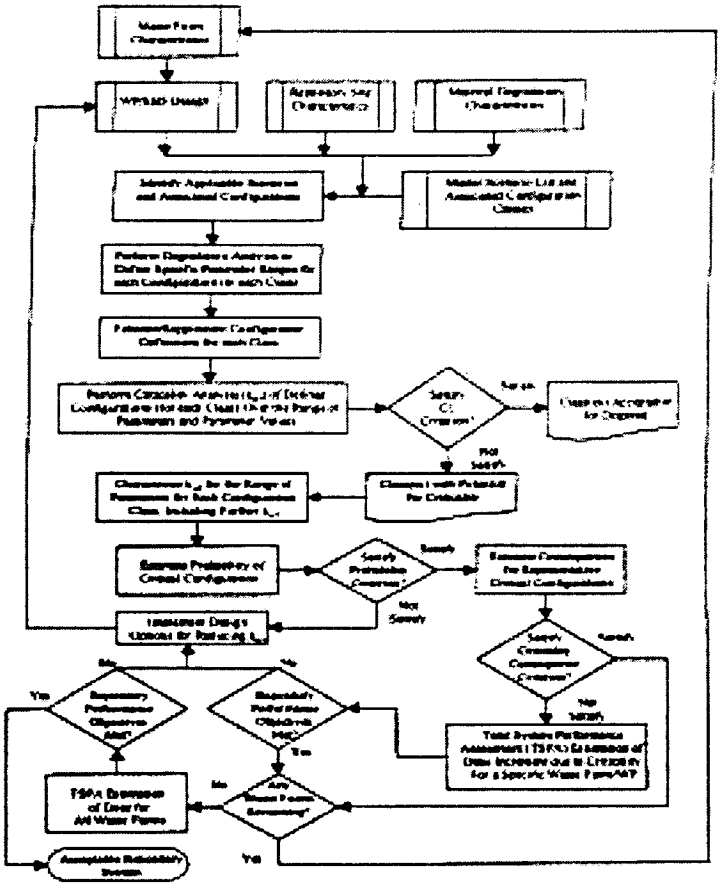

**Figure 3.1-1 Overview of disposal criticality analysis methodology**

However, the staff does not expect major changes in the proposed Part 63 requirements with respect to allowing the use of probabilities and consequences to assess the repository performance.

10 CFR 63.113 and 63.114 provides the performance objective and requirements for the repository after permanent closure. These two sections in Part 63 provide the applicant with the flexibility to take into account the probability and/or consequences of the events as **one of the factors** in designing the criticality control system of the WP. On the other hand, the multiple-barrier requirement in 10 CFR 63.113 indicates that total reliance on a probabilistic approach is not acceptable to the NRC. This type of requirement in Part 63 makes the approach risk-informed as opposed to risk-based.

Based on the description provided in Section 1.5 of the TR as revised in the responses to the RAI, the NRC agrees that DOE's proposed approach, as described in Figure 3-1 of Attachment B, in the responses to the RAI, is a risk-informed approach and acceptable, provided:

- Consequence analyses shall be performed for the critical configurations whose probabilities are smaller than the Probability Criterion established for design modifications as indicated by Figure 1-1.

- The cumulative dose increment for all those critical configurations that did not satisfy the Consequence Criterion but met the Repository Performance Objectives shall be included in the TSPA estimation of dose for all waste forms as indicated by Figure 1-1.

### 3.2 Design Criteria

The four design criteria shown in Figure 3.1-1 as decision points in the diamond-shaped boxes represent the design criteria for which the DOE is seeking approval. The DOE's request with respect to design criteria in Enclosure 2 of the RAI responses is:

> *"DOE requests acceptance of the four design criteria presented in Section 1.2, Part A of the Topical Report as acceptable for ensuring that design options are properly implemented for minimizing the potential for, and consequences of, criticality. The design criteria are discussed in Sections 3.4, 3.5, 3.6, and 3.7."*

The following sections provide the DOE's specific requests with respect to each of the four design criteria, the basis for request, the basis for staff evaluation, and the outcome of the evaluations. The basis for staff evaluation for the following four design criteria is the third AC stated in Section 2.3.2 ("Container Life and Source Term IRSR"). The use of Regulatory Guides and Standards are mentioned in Section 2.4 and described in the following pertinent subsections.

### 3.2.1 Critical Limit (Sections 1.2A.1, 3.4.3, 3.4.4 of TR)

In Section 1.2, Item A.1., the DOE has requested that the following design criterion be accepted.

> *The Critical Limit (CL) criterion discussed in Section 3.4: the calculated $k_{eff}$ for systems (configurations) for postclosure will be less than the CL. The CL is the value of $K_{eff}$ at which the system is considered potentially critical as characterized by statistical tolerance limits.*

Section 3.4.4 presents the discussion on the CL. The CL is placed on the calculated value of $k_{eff}$ at which a configuration is considered potentially critical and accounts for the criticality bias and uncertainties. The discussion and the staff evaluation of criticality bias and uncertainties are presented in Section 3.5.3.2 of this SER, as part of how the CL is determined.

Subject to its appropriate implementation in the LA, the staff agrees with the concept of establishing a $k_{eff}$ limit that would include all the appropriate biases and associated uncertainties for each in-package and out-of-package configurations.

### 3.2.2 Criticality Probability (1.2.A.2, 3.5)

In Section 1.2 of the TR, the DOE has requested an approval for:

> *The Probability criterion discussed in Section 3.5: a criticality frequency of $10^{-4}$ per year for the entire repository will not be exceeded in any of the first 10,000 years for all combinations of waste packages and waste forms. This criterion is intended to ensure that the expected number of criticalities is less than one during the regulatory life of the repository (10,000 years). It is used to define a waste package criticality control design requirement in support of defense-in-depth with respect to the Repository Criticality Performance Objective in item 4* (Item four being the Performance Objectives Criterion).

Section 3.5 of the TR describes the part of the methodology related to estimating the probability of criticality occurring inside the WP. It states that the first of the two objectives of the probability calculations is to estimate the risk of criticality in terms of the overall increase in radionuclide inventory and the effects on dose at the accessible environment. The second objective is to provide an estimate of the effectiveness of the variety of measures used to control or limit postclosure criticality. However, the DOE states that the "... derived probability criterion is not for regulatory purposes, and will be used only to guide decision processes internal to waste package design."

DOE further indicates that the calculated probability of criticality per WP is compared with the WP criticality probability criterion, derived from the repository criticality

probability criterion. The repository criticality probability criterion is defined as an expected number of criticalities of less than one in 10,000 years. This value is the one assumed during the regulatory period.

In addition, DOE lists factors that could complicate the allocation of the repository criticality probability criterion to a per-WP and per-year basis. Nevertheless, the DOE has established the WP probability criterion of approximately $10^{-4}$ expected criticalities, per waste package, in 10,000 years.

Proposed 10 CFR 63.114; AC 5, in the TSPA IRSR; AC 3, in the CLST IRSR; and AC 1, in RT IRSR, were used in evaluating the criticality probability criterion.

With respect to the design selection, it is not appropriate for the staff to make any evaluation or provide any judgement. With respect to using the criticality probability criterion in the overall methodology, the result of the evaluation is provided in the following.

As acknowledged by the DOE, there are some factors that the DOE has not considered, and these factors would increase the probability of criticality per WP. Some of these factors are:

- The probabilities of WP breach and loss of neutron absorber increase with time.
- There is a possibility of common mode failure. In particular, for external criticality, there is a possibility of multiple packages contributing to the accumulation of fissionable material at a single location.

Nevertheless, examination of Figure 1-1, in the TR, as revised in Attachment B of the RAI responses and as shown in Figure 3.1-1 of this document, indicates that consequence analyses shall be performed regardless of the value for the probability of criticality per WP. However, Figure 3.1-1 indicates that the probability comes into play when calculating the expected dose (i.e., probability-consequence product). Therefore, the staff does not agree with the DOE that "this derived probability criterion is not for regulatory purposes..."

In addition, in Section 3.6 of the TR, "Estimating Consequence of Criticality Event", the DOE states that "... when the $k_{eff}$ of the configuration analyzed exceeds the CL and the probability of occurrence of that configuration exceeds the waste package probability criterion, currently derived in Section 3.5 as approximately $10^{-4}$ per waste package in 10,000 years, a consequence analysis is performed." However, in its comments (U.S. DOE, Comments on Draft SER, May 2000), on the draft version of this document, the DOE has indicated that there is a misstatement in Section 3.6 of the TR. In its comments DOE indicates that when the $k_{eff}$ of the configuration analyzed exceeds the CL and the probability of occurrence of that critical configuration is **below** the waste package probability criterion, a consequence analysis is performed.

The staff believes as long as the probability criterion, described in Sections 1.2.A.2 and 3.5 of the TR, is used merely for deciding among different design options for the criticality control systems in WPs, the staff has no objection. However, the staff stresses that only the probability defined in the proposed 10 CFR 63.114(d) can be used for screening the criticality events.

### 3.2.3 Criticality Consequence (1.2.A.3, 3.6.3)

The DOE has proposed to establish design criteria with respect to criticality consequence. The purpose of establishing criticality consequence design criteria is "... to get rapid design guidance feedback without lengthy TSPA calculations." In other words, the impact of criticality on the dose to critical group for those configurations that have consequences less than some limit will not be analyzed. Section 1.2A.3. of the TR requests the following design criterion be accepted.

> *The Criticality Consequence criterion discussed in Section 3.6: the expected radionuclide increase from any criticality event will be less than 10 percent of the radiologically significant radionuclide inventory (curies present at time of criticality) that is available for release and transport to the accessible environment. This criterion is intended to ensure that the average radionuclide increment from any single criticality is much less than the uncertainty of the performance assessment dose estimation, and is also used to define a waste package criticality control design requirement in support of defense-in-depth with respect to the Repository Performance Objective in item 4.*

The DOE has proposed to use a radionuclide inventory increase limit for screening those steady-state criticality events that would not add any significant dose to the total expected dose to the critical group. The DOE's limit pertains only to a steady-state criticality and includes the radionuclide increase as the only consequence. The DOE has selected the value for the criticality consequence criterion "... to be one order of magnitude less than the uncertainty of a typical input parameter to the TSPA estimate of dose." In addition, the DOE has stated that the criticality consequence criterion is intended for design guidance only. In any case, the revised Figure 3.1-1 indicates that a TSPA estimation of incremental dose from any radionuclide increase from steady-state criticality will be included in the full TSPA for the repository.

The staff used AC 6 and 12 from the TSPAI IRSR, and AC 3, in the CLST IRSR, for evaluating the DOE's proposed criticality consequence criterion.

As acknowledged by the DOE in the response to RAI 3-23, an increase in the radionuclide inventory is not the only consequence of a steady-state criticality. Therefore, using the radionuclide inventory from a steady-state criticality as an indication for no significant impact to repository information may not be entirely true.

However, the DOE has indicated in the response to RAI 3-23 that "... when the radionuclide increment and risk are calculated, the time dependencies of temperature and power will be incorporated into the calculations." Furthermore, the consequence criterion with respect to transient criticality needs to be identified. Therefore, the staff agrees that a 10 percent increase in radionuclide inventory from internal steady-state criticality can be used as a consequence criterion for deciding whether to perform a TSPA analysis or to bypass it during the design process, provided that:

- The consequences of a 10 percent radionuclide increase and associated heat production are determined.
- The heat impact and other degradation impacts of steady-state criticalities are included in the full TSPA analysis.

If the DOE is selecting a criterion for one of the consequences of criticality, it is not clear why criteria for other types of criticalities are not established. Other types of criticalities include internal transient and external steady state.

**Open Item**

- The consequence criteria for transient and external criticalities are not addressed in the TR. The DOE must specify if it intends to perform full consequence analyses for transient and external criticality events and include them in TSPA or use some type of criteria for the purpose of criticality control design selection.

### 3.2.4 Performance Objectives (A.4, 3.7, 4.5)

Under the design criteria, the Repository Performance Objectives Criterion is proposed for acceptance in Section 1.2 of the TR. The DOE has requested that the following design criteria be accepted:

> *"The Repository Performance Objectives criterion discussed in Section 3.7: the ability to satisfy dose rate performance objectives will not be compromised by the radionuclide increment due to criticality events (if any)."*

In Section 3.7, "Estimating Criticality Risk," the DOE states that "... the purpose of this section is to summarize the role of criticality in the performance assessment process for illustrative purposes; acceptance of the performance assessment methodology, per se, is the subject of other documents."

The section also discusses the approach to estimate the potential increase in dose at the accessible environment, and the approach to incorporate the results into the TSPA. For a repository criticality, it defines risk as the product of probability and consequence, and states that the consequence will be measured, in practice, by a parameter with

significant health impact, such as radiation dose to the nearby population. TSPA will be used to estimate the radiation doses.

The staff evaluation of the methodology for estimating the risk from criticality events will be performed in Section 3.8. In this section of the SER, the staff's evaluation is concentrated on the adequacy of the proposed repository performance criteria.

The staff used the proposed 10 CFR 63.113 for the evaluation of the DOE proposed Repository Performance Criterion.

The staff agrees that the impact of postclosure criticality events on the repository should be assessed using an overall performance criterion that has been specified in the proposed 10 CFR 63.113 (b) as the expected annual dose to the average member of the critical group not exceeding 0.25 mSv (25 mrem) TEDE at any time during the first 10,000 years after permanent closure, as a result of radioactive materials released from the geologic repository. The staff further agrees that the ability to satisfy the dose rate performance objectives must not be compromised by the radionuclide increment caused by criticality events. However, in its overall evaluation, the DOE must:

(a) Consider all aspects of criticality events including, but not limited to, increase in radionuclide, waste heat output, and degradation of EBS.

(b) Define what is considered to be an insignificant impact.

(c) Include all the probability-consequence products from postclosure criticality into the full TSPA, as indicated by Figure 1-1 and Section 3.6.3, last sentence in TR.

## 3.3 Criticality Scenarios

### 3.3.1 Internal Criticality Scenarios (Sections 3.1, 3.1.1)

The DOE has proposed a Master Scenario List (MSL) that consists of a standard set of degradation scenarios that must be considered part of the criticality analysis of any waste form (WF) disposed of in the repository. Section 3.1 of the TR provides a description of the internal scenarios that will be evaluated for criticality. In its responses to the RAI, the DOE has requested approval of the MSL as stated by the DOE in the following paragraph:

> *DOE requests acceptance that the list of standard scenarios outlined in Figures 3-1 and 3-2, as supplemented by the new Sections 3.1.3 and 3.1.4 to be added to the Topical Report as discussed in the response to RAI 3-1, comprehensively identifies the generic degradation scenarios incorporating those features, events, and processes associated with the*

*proposed repository at Yucca Mountain that may significantly affect the potential for, and consequences of, criticality.*

The development of degradation scenarios is based on a combination of FEPs, within the YMR, that result in degraded configurations to be evaluated for criticality. Groups of similar degraded configurations are combined into configuration classes to reduce the calculational burden while ensuring that a comprehensive set of configurations is considered. These configuration classes consist of configurations with similar material compositions and geometries that differ because of parameters, such as uranium enrichment and burnup or water infiltration rate, which vary over a given range.

The internal scenarios are combinations of FEPs that may result in critical configurations inside the WP and are determined based on several discriminators. The top-level discriminator is whether the location of the initial WP penetration is at the top or bottom of the package, which determines whether water can accumulate inside the package. The second level discriminator is the rate of degradation of the WP internal structures as compared with the degradation of the WF. Lower-level discriminators include items such as the transport characteristics of the fissile materials (FMs) and structural materials. These scenarios can result in the following configuration classes, which the DOE has identified as having the potential to support a criticality event and, therefore, requiring analysis for all WFs:

- The WP internals are degraded, but the WF remains relatively intact. Criticality is possible if water fills the lower portion of the package, neutron absorber is flushed from the WP, and little fissionable material is removed from the package.

- Both the WP internal structure and WF are degraded and resting on the bottom of the WP. There is a criticality potential if water fills the lower portion of the package, absorbing material is flushed out of the package, and most of the fissionable material remains within the WP.

- The WP internals remain relatively intact, but the WF is degraded. A criticality event could be possible if water fills the lower portion of the package and the degradation of the WF causes physical separation between the fissionable material and the neutron absorber.

- The WF accumulates at the bottom of the WP along with water trapped in clay or with hydrated corrosion products without any standing water in the package.

- The fissionable material is trapped along with water in clay filling the WP and distributed throughout the WP volume.

- The WF degrades in place with the WP internals intact. No additional separation is created between the fissionable materials and the neutron absorbers in the WP internal components, but the degraded WF is more reactive than the intact WF for some fuel types.

The DOE, per response to RAI 3-1, has addressed how disruptive events such as seismicity and volcanism will be evaluated in the MSL. Seismic events will be addressed by identifying representative seismic predecessor configurations that could be transformed to one of the six previously identified critical configurations by a seismic event. These critical configurations are distinguished by the level of degradation of the basket and the WF. The predecessor configurations will have significantly higher gravitational potential energy than the critical configurations, such that a seismic event could rearrange the materials to form critical configurations. The seismic event could increase the reactivity of the system by shifting fuel assemblies such that more assemblies fall below the water level in the WP, causing individual fuel pins to collect in a more compact configuration, or moving poison or moderator-excluding materials such as the basket structure of corrosion products away from the fuel and replacing them with water. A critical configuration created by a seismic event would likely lead to a transient criticality because of the relatively short period of time that reactivity is inserted. The probability of the occurrence of a transient criticality initiated in this manner would be determined by considering both the probability of occurrence of the predecessor configuration and the probability of a seismic event of sufficient magnitude occurring to take such configurations to criticality. The probability and consequences of the transient criticality will be used to determine a transient criticality risk caused by seismicity.

The NRC staff reviewed the MSL against the AC in the IRSRs to determine whether all FEPs that have the potential to increase the reactivity of the system inside the WP can be evaluated in the identified scenarios (CLST AC No. 4) or can be eliminated on the basis of low probability (TSPAI AC No. 11) or low consequence (TSPAI AC No. 12). Additionally, staff review determined whether a comprehensive list of processes and events have been identified (TSPAI AC No. 1) and whether adequate documentation was provided identifying how the initial list of processes and events were grouped into categories (TSPAI AC No. 2). Staff review was supplemented by examination of a preliminary DOE FEPs database (U.S. Department of Energy, 1999) to ensure that the FEPs identified as important to the criticality evaluation inside the WP in the FEPs database can be accounted for with the MSL.

The NRC staff found that grouping sets of similar configurations into configuration classes is a reasonable way to reduce the calculational burden but still provide reasonable assurance that the probability of criticality will not be significantly underestimated. The NRC staff found that the MSL and the additional analyses conducted for seismic events, as stated in the response to RAI 3-1, will adequately identify scenarios that may significantly impact the potential for, or consequences of, a criticality event within the WP, based on the FEPs associated with the YM repository.

The NRC staff found that all FEPs important to the criticality evaluation inside the WP can be incorporated in the proposed methodology if the additional steps accounting for seismic events are performed.

### 3.3.2 External Criticality Scenarios (Sections 3.1, 3.1.2)

A component of the DOE request discussed in Section 3.3.1 is acceptance of the list of external scenarios in Figure 3-2 of the TR. External scenarios are combinations of FEPs that may result in critical configurations outside the WP. These scenarios, classified as NF or FF, depending on their location relative to the drift walls, are preceded by one or more internal scenarios (Section 3.3.1 of this SER) that result in release of FM from the WP. (There is one exception, scenario NF–5.) Simplified descriptions of the DOE scenarios follow; note that the invert is assumed to contain concrete, crushed tuff, or both.

NF–1: solute transport of FM from the WP and accumulation in the invert.
NF–2: slurry transport of FM from the WP and accumulation on the invert.
NF–3: colloidal transport of FM from the WP and accumulation in the invert.
NF–4: water ponds in drift, WP and WF degrade, and FM accumulates in clays at the bottom of the drift.
NF–5: water ponds in drift, WP degrades, and intact WF sits in pond.

FF–1: solute transport of FM from the drift and chemical accumulation in the unsaturated zone.
FF–2: colloidal transport of FM from the drift and accumulation in the unsaturated zone.
FF–3: solute transport of FM from the drift and chemical accumulation in the saturated zone.

All scenarios require—in addition to release and transport of FM—a degree of separation of FM from neutron absorbers, and mechanisms for this process are therefore included. Each scenario encompasses one or more configuration classes, which further specify the processes and setting that define the potentially critical configuration. For example, scenario NF–3 includes three configuration classes—NF–3a, NF–3b, and NF–3c—that specify whether colloids accumulate in WP corrosion products, invert fractures, or degraded concrete, respectively.

The DOE constructed and validated this list of external scenarios and configuration classes as part of the TSPA—Viability Assessment (TSPA-VA) abstraction effort, and was thus informed by the comprehensive, site-specific TSPA scenario analysis (U.S. Department of Energy, 1998). The list was produced at a DOE workshop and reviewed by a group of experts. The expert review constitutes the DOE validation of the process of identifying scenarios.

The DOE provided additional, clarifying information in its responses (Letter, November 19, 1999) to NRC RAI items 3-2, 3-4, 3-5, 3-6, 3-7, and 3-8, that resolved NRC questions on external scenarios.

- RAI 3-2. In response to the NRC comment that the DOE should include external scenarios with potentially positive neutronic feedback characteristics, the DOE stated that such configurations (a more appropriate term than scenarios) are explicitly included in the TR discussion of criticality consequence modeling (Section 4.4.1.2).

- RAI 3-4, 3-5, 3-6, 3-7, and 3.8. These items dealt with either: (i) inappropriate wording, in the TR, that gave the erroneous impression that certain processes or mechanisms were excluded when configuration classes were defined; or (ii) editorial ambiguities. For example, RAI 3-5 pointed out that the TR appeared only to be considering reduction as a mechanism for FM precipitation. The DOE will change the TR to reflect the fact that reduction is merely an example of the possible mechanisms.

Staff assessed the comprehensiveness of the DOE scenario list with reference to NRC and DOE documents concerned with FEPs external to the WP. In particular, NRC IRSRs on the evolution of the NF environment (ENFE) (U.S. Nuclear Regulatory Commission, 1999a) and RT (U.S. Nuclear Regulatory Commission, 1999b) KTIs provided extensive discussions on the range of FEPs that may affect the fate and transport of contaminants, including FMs and neutron absorbers, in the NF and FF. This assessment applied AC from these two IRSRs, as well as from the TSPAI IRSR (U.S. Nuclear Regulatory Commission, 2000), listed in Section 2.3 of this SER. From Section 2.3.1 (TSPAI), relevant criteria are numbers 1, 2, 3, 7, and 8, concerned with the establishment and categorization of a scenario list. From Section 2.3.3 (ENFE), criteria relevant to scenarios are 14 and 15. From section 2.3.4 (RT), criteria 1 was applied. Staff review was supplemented by examination of a preliminary DOE FEPs database (U.S. Department of Energy, 1999a). Specifically, the database was checked for: (i) inclusion of all external scenarios delineated in the TR; and (ii) inclusion of any external scenarios not in the TR.

The occurrence of igneous activity in the repository would lead to a configuration significantly different from those identified in the MSL, so the DOE will provide a completely separate analysis to determine the criticality potential from a volcanic event. This analysis, as described in response to RAI 3-1, will consist of a probabilistic methodology to determine the probability of a criticality event after an igneous intrusion. The analysis will consist of the following four steps: (i) evaluating the probability of the WP breaching during a volcanic event; (ii) modeling the transport of fissionable material after breach of the WP; (iii) determining the accumulation of FM from the magma flow; and (iv) determining the potential for criticality of any accumulations identified using silica, water moderation, or both, as appropriate.

If the probability of criticality resulting from igneous activity exceeds the TSPA screening level of $10^{-8}$/yr for the entire repository, the consequences of these criticality events will have to be determined to evaluate the risk associated with such an event.

NRC staff found that—contingent on the TR revisions promised in the DOE RAI responses and on addressing an additional scenario described in the next paragraph—the external MSL, as depicted in Figure 3-2 of the TR and described in Section 3.1 of the TR, comprehensively identifies generic, site-specific scenarios that may impact significantly the potential for, and consequences of, a criticality event external to the WP at YM. The scenarios cover the range of reasonably conceivable physical and chemical processes that could affect the external disposition of FM and neutron absorbers. The integration of the criticality scenario analysis with the TSPA scenario analysis lends confidence to the criticality effort because the DOE increasingly is making scenario analysis a transparent process that the NRC can readily evaluate. The criticality scenario list should be flexible enough to respond to any future changes in the TSPA scenario analysis. For example, design changes affecting the composition and dimensions of the invert will affect the viability and completeness of the scenarios involving invert accumulation of FM. NRC staff found the framework that the DOE presented in the response to RAI 3-1 to analyze the probability of igneous-activity-induced criticality is acceptable. The response to the RAI, however, does not present a sufficient level of detail about how each of the four steps in the analysis will be conducted to make a finding of whether the analysis will be considered appropriate by the NRC staff.

One potential exception to the comprehensiveness of the TR scenario list was identified in the DOE FEPs database (U.S. Department of Energy, 1999a). Database entry 2.2.14.07.00 concerns FM precipitation caused by dryout in a perched water basin, and is described as an "alteration" of configuration class FF–1a. However, in its comments (U.S. DOE, May 2000), the DOE stated that "DOE would add a specific reference to "dryout" in the more general discussion of perched water precipitation in configuration 9."

**Open Item**

- The DOE needs to provide a modeling approach for igneous-activity-induced criticality.

## 3.4 Criticality Configurations

### 3.4.1 Internal Configurations (Section 3.2)

The determination of internal configurations is based on the parameter ranges of the variables that describe the potentially critical internal configurations. In its responses to

the RAI, the DOE has requested an approval of the methodology for determining internal configurations, as stated by the DOE in the following:

> *DOE requests acceptance of the method for generating a comprehensive set of potential postclosure configurations as discussed in Sections 3.2 and 3.3 of the Topical Report. The principal components of this method are given in the following list.*
>
> - *Degradation methodology: Ability of the methodology to calculate the loss of fissionable elements and neutron absorbers and calculate the composition of degradation products precipitating in the waste package. For this purpose we are requesting acceptance of the use of a steady-state geochemistry code. Improvements are still being incorporated into this methodology. The example discussed in the Topical Report (EQ3/6 in pseudo-flow-through mode, discussed in Section 4.2.2), has been replaced by the solid-centered-flow-through EQ3/6 code discussed in the response to RAI 4-32, which includes proposed modifications to Section 4.2.2. Use of this code is planned to be demonstrated and justified to support licensing.*
>
> - *Configuration generator: (1) Use of time-dependent first-order differential equations, solved by numerical integration, to track the concentration, or amount of fissionable or neutron absorber material (Section 4.3.3); (2) Development of coefficients or terms of these equations by abstraction from steady-state geochemistry code calculations (Section 4.3.4); and (3) Random variation of terms or coefficients in these equations as part of a Monte Carlo calculation to reflect the uncertainty in the rates and location of natural processes (Section 4.3.4). In implementing #2, the appropriate balance between the use of EQ3/6 and the Configuration Generator Code will be demonstrated for each major waste form category as part of the License Application process. Examples of the use of #3 are given in Appendix C of the Topical Report, Sections 3.3 and 4.1, in response to RAI 4-38, and in CRWMS M&O, Probability of Criticality for MOX SNF, CAL-EBS-NU-000007.*
>
> - *DOE requests acceptance of the validation process for the degradation analysis methodology that uses the solid-centered-flow-through mode (an improvement on the pseudo-flow-through mode described in Section 4.2.2 of the Topical Report, as discussed in the response to RAI 4-32). DOE also requests acceptance for the validation process for accumulation*

*methodology that uses a geochemistry-transport code or a geochemistry code used in a mode that simulates transport. This validation is expected to be provided by comparison between codes, comparison with experimental data, and comparison with natural analogs. These comparison cases are summarized in Tables 4-2 and 4-3 of the Topical Report. We do not seek acceptance of the bounding cases, which have been identified for the current range of environmental parameters and may be modified for the environmental parameters applicable to the Yucca Mountain repository that will support licensing.*

#### 3.4.1.1 Internal Configuration Methodology (Section 3.2)

The proposed methodology to determine internal configurations is to use the appropriate range of configuration parameters to further specify the identified configuration classes for each combination of WF and WP. This methodology will be accomplished by performing a geochemical analysis for each configuration class to identify the chemical composition of the corrosion products remaining in the WP and by determining the physical properties of the remaining corrosion products. This end result will be a specific and detailed range of configurations that must be considered in the parametric criticality evaluation of each configuration class.

The geochemical processes will be used to track the location of important fissionable, neutron-moderating, and neutron-absorbing materials, and will be specified using the following steps:

1. Identify specific corrosion rates for all internal components, the range of drip rates of water onto a WP under a dripping fracture, and the range of dripping-water chemistry parameters.

2. Estimate the location of potentially reacting materials to determine if a reaction is possible.

3. Perform probabilistic flow-through mode geochemical calculations for the representative parameter range for each configuration class.

4. Determine concentrations of fissionable materials and neutron absorbers in solution and in solids and insoluble corrosion products within the package.

5. Determine whether clay has formed from chemical alteration of glass WFs or from the silica and alumina in the water, and determine the amounts of undegraded material and solid degradation products present.

6. Determine the range of hydration of degradation products possible if the package is not flooded.

7. Quantify the amounts of undegraded material and solid degradation products present for each configuration class.

8. Evaluate the potential for adsorption of soluble FM or of neutron-absorbing material on corrosion products.

At appropriate intervals in the progress of the geochemical process, physical processes will be evaluated. These physical processes include possible locations for solids; the density and physical stability of corrosion products; the thermal and structural behaviors of the internal structures and the WF; and the effects of external events on the internal components, WF, or the location of the corrosion products.

The DOE, per response to RAI 1-4, has indicated it will evaluate the probability of occurrence of all configurations identified as potentially autocatalytic in published articles. This evaluation will provide additional confidence that all realistic potentially high-consequence criticality events have been considered.

The NRC staff reviewed the methodology that the DOE will use to identify critical configurations against the acceptance criteria in the IRSRs to ensure that the proposed methodology will identify the configuration classes and configurations that have potential for nuclear criticality (CLST AC No. 5). The methodology uses models to develop the configurations of interest, but acceptance of these computer codes is beyond the scope of the review of the TR. Modeling verification and validation of these computer codes will be evaluated when the DOE submits the appropriate validation reports.

The NRC staff found that, provided the DOE evaluates the probability of occurrence of all configurations identified as potentially autocatalytic in published articles, as stated in the response to RAI 1-4, the proposed methodology is sufficient to provide reasonable assurance that the analysis has been performed on a comprehensive set of internal configurations and that no configuration that could increase substantially the calculation of the probability or consequence of a criticality event has been omitted from the analysis.

### 3.4.1.2 Internal Configuration Modeling Approach (Section 4.2)

The determination of internal criticality configurations depends on the degradation rate of WP barrier materials, internal components, and WFs determined from the quantity of water contacting the material and the chemistry of the dripping water.

Per response to RAI 4-25, individual corrosion models are developed based on data from the DOE's material testing program for each of the materials that make up the WP and WF. WP degradation models will be the models used in the TSPA that output a distribution of breach times at various locations on the WP for a given set of environmental conditions. The degradation rates used in the criticality evaluation also will be consistent with the WF corrosion models used for TSPA.

The geochemistry within the WP will be calculated using a commercial software code such as EQ3/6 (Wolery, 1992). The software will be qualified under an appropriate QA program. A series of runs of the geochemistry code will be used to simulate water dripping into and leaking out of a WP. In response to RAI 4-32, the DOE stated that a modification to the EQ6 (Wolery and Daveler, 1992) portion of the code called the solid-centered-flow-through code, will be used to model water inflow and outflow and track the time step adjustment.

The configuration generator code will be used to track the concentrations of neutronically significant isotopes and chemical species that can affect the solubility of the neutronically significant elements. This code uses time-dependent, first-order differential equations to represent the chemical transformations of elements or compounds that have coefficients determined by fitting data from detailed calculations of a geochemistry code such as EQ3/6 (Wolery, 1992). The code will provide bookkeeping for the transport between sites of the application of a detailed geochemistry code and, in some situations, provide more rapid calculation where the detailed geochemistry code results can be used to develop heuristic models for the most significant ions for a few solution parameters.

At each time step, the configuration generator code will calculate the increase in the quantity of water in the WP; the amount of each element dissolved in this water; the amount of each element lost because of the removal of water from the WP; the pH of the water; the solubility of materials in the water inside the WP; and the precipitation or dissolution of the species being tracked, based on solubility. Uncertainties in parameters that will be used in these equations will be represented using the Monte Carlo technique. These uncertain parameters will be assigned distributions of possible values. Many realizations will be conducted by sampling a single value from the distributions of values assigned to all the uncertain parameters and calculating the results for each realization. The use of the Monte Carlo technique is fully described and evaluated in Section 3.6 of this SER.

The NRC staff reviewed the methodology that the DOE will use to identify critical configurations against the acceptance criteria in the IRSRs to ensure that the proposed methodology, including the degradation models, geochemistry codes, differential equations used to track locations of materials, and the coefficients that will be used with these differential equations, will identify the configurations that have potential for nuclear criticality (CLST AC No. 5). The methodology uses models to develop the

configurations of interest, but the acceptance of these computer codes is beyond the scope of the review of the TR.

The NRC staff found that using corrosion models consistent with those used in the TSPA, for the license application for YM, was acceptable for determining breach times of the WP, and degradation rates of the WF and other components inside the WP. These models will be reviewed in detail during the review of the performance assessment. The DOE only needs to demonstrate that these models do not contain any assumptions that are conservative for performance assessment calculations, but non-conservative for the criticality analysis. The NRC staff found the use of a steady-state geochemistry code with modifications to track the quantity of water in the WP is acceptable to calculate the loss of fissionable elements and neutron absorbers and the composition of degradation products precipitating in the WP, as long as the code is properly validated and verified. Additionally, NRC staff found the use of differential equations is acceptable to track the concentration of fissionable and neutron-absorbing materials, as long as the coefficients for these equations are developed based on sufficient and appropriate data. NRC staff found the abstraction of the results from a steady-state geochemistry code is acceptable to develop the coefficients for these equations. Findings on the use of Monte Carlo calculations to simulate the uncertainty in the rates and locations of natural processes are found in Section 3.6 of this SER.

#### 3.4.1.3 Internal Configuration Validation Approach (Section 4.2.4)

The DOE proposes to not validate models that have been validated and used in the TSPA because the model validation will be evaluated during the LA review process for the repository. The degradation rates of internal components not modeled and validated in the TSPA will be developed from material test data and will be validated based on information and data provided as part of the disposal criticality analysis supporting the LA.

The geochemical code modified to track the water movement of, and used to determine the chemical environment inside, the WP, will be compared against analytical solutions and against results obtained by chaining several thousand individual EQ6 runs, with adjustment of the water mass between runs. Additionally, the geochemical code will be validated by comparison with the other geochemistry-transport codes. The validation of specific computer codes is beyond the scope of this review, so no finding will be made as to the acceptability of the use of the EQ3/6 code for the repository environment.

The NRC staff reviewed the proposed methodology that the DOE will use to validate and verify the computer codes, used to identify critical configurations, against the acceptance criteria in the IRSRs. The methodology was reviewed to ensure that the approach to model validation and verification for the degradation models and geochemical codes will provide confidence that the codes will provide a reasonable representation of the configuration classes and configurations, that may occur in the repository, with a potential for nuclear criticality (TSPAI AC No.16 and AC No.18). The

acceptance of these computer codes is beyond the scope of the review of the TR. Modeling verification and validation of these computer codes will be evaluated when the DOE submits the appropriate validation reports as part of the LA.

The NRC staff found that the proposal to not validate models that have been validated and used in the TSPA is acceptable, as long as the model is being used for the same purpose for which it was used in the TSPA, and no assumptions were made, in the TSPA modeling, that were conservative for purposes of performance assessment, but could be nonconservative for criticality analyses. The NRC staff will review the corrosion data supplied by the DOE and used as input in the model calculations during the review of the DOE TSPA. The proposed methodology of validating and verifying the geochemistry code used to determine the chemical environment inside the WP by comparing: (a) the results of the code to analytical solutions; (b) results obtained by chaining several thousand individual code runs; and (c) results obtained using other geochemistry-transport computer codes, is acceptable to the NRC staff. The acceptability of the EQ3/6 to model conditions expected in the repository has not been evaluated in this review, however. Additionally, the NRC staff found that the proposed methodology for validation of the configuration generator code by comparing the results of the code with appropriate hand calculations is acceptable. In the LA, the DOE will have to provide verification that all computer codes used in the analysis are being implemented correctly and demonstrate that using these computer codes does not underestimate the probability of a criticality event for all WFs that will be disposed of in the repository. The NRC staff will review the corrosion models as well as the corrosion data provided by the DOE and used as input in the model calculations, to assure that these are acceptable and an appropriate correspondence exists between the TSPA calculations and those used in the criticality analysis.

### 3.4.2 External Configurations (Section 3.3)

The determination of external configurations is based on the parameter ranges of the variables that describe the potentially critical external configurations. In its responses to the RAI, the DOE sought approval of the methodology for determining external configurations as part of the following request:

> *DOE requests acceptance of the method for generating a comprehensive set of potential postclosure configurations as discussed in Sections 3.2 and 3.3 of the Topical Report.*

Two of the three principal components listed with this request are pertinent to external configurations:

- *Configuration generator: (1) Use of time-dependent first-order differential equations, solved by numerical integration, to track the concentration, or amount of fissionable or neutron absorber material;*

*(2) Development of coefficients or terms of these equations by abstraction from steady-state geochemistry code calculations; and (3) Random variation of terms or coefficients in these equations as part of a Monte Carlo calculation to reflect the uncertainty in the rates and location of natural processes.*

- *Accumulation Methodology: Ability of the methodology to calculate the accumulation of fissionable elements external to the waste package. For this purpose we are requesting acceptance of the use of a geochemistry-transport code and/or a geochemistry code used in a mode that simulates transport.*

In addition, the DOE in the RAI response requested acceptance for the approach to validating the configuration models in the following excerpts:

- *DOE also requests acceptance for the validation process for accumulation methodology that uses a geochemistry-transport code (e.g., PHREEQC, described in the response to RAI 4-32) or a geochemistry code used in a mode that simulates transport (e.g., EQ3/6 in the open-system mode described in Section 4.2.3).*

- *We also request acceptance that the configuration generator models described in Sections 4.3.3 and 4.3.4 can be validated by appropriate hand calculations.*

As discussed in Section 3.3.2, the external criticality scenarios are grouped into configuration classes that define the particular sequence of FEPs that lead to a potentially critical configuration outside the WP. Within a configuration class, individual configurations reflect specific choices of parameters that define the input to criticality models. Staff assessment of the DOE external configuration methodology is addressed in three subsections, concerned with methodology, modeling, and validation approach.

### 3.4.2.1 External Configuration Methodology (Section 3.3)

In Section 3.3 of the TR, the DOE briefly describes how it will develop these external configurations by quantifying parameter ranges for the configuration classes. Formulation of a configuration is based on parameters consistent with repository features, taking into consideration current design and site characterization. Examples listed include drift floor materials and host rock fracture density. The six steps for formulating a configuration are determinations of:

- An FM source term using information from generation of internal configurations (Section 3.4.1 of this SER);

- Water flow rates and patterns;

- Sorption along flow paths;
- Mineral precipitates along flow paths;
- Alternate paths when primary rock fractures are filled, including possible coalescence of contaminant plumes from several WPs;
- Reaction products resulting from the plume encountering a reducing zone.

The technical basis for the overall approach to identifying configurations is not addressed explicitly, but it is tied to modeling and validation discussed elsewhere in the TR and reviewed in Sections 3.4.2.2 and 3.4.2.3 of this SER.

Staff assessed the DOE method for generating external configurations with reference to NRC and DOE documents concerned with mechanisms of radionuclide release and transport in the repository environment. In particular, NRC IRSRs on the ENFE (Nuclear Regulatory Commission, 1999a) and RT (U.S. Nuclear Regulatory Commission, 1999b) KTIs provided extensive discussions on the range of FEPs that may affect the fate and transport of contaminants, including FMs and neutron absorbers in the NF and FF. AC from these IRSRs relevant to configuration methodology are Section 2.3.3, items 3, 4, 14, and 16 (ENFE); and Section 4.4.1, items 1, and 2 (RT). Some of these criteria refer to specific choices of parameters and models—which are beyond the scope of this review—but the methodology was reviewed for the likelihood that it will address these criteria. There are other AC (e.g., concerned with mathematical modeling) that will need to be addressed by the DOE criticality analysis when it is performed for the LA. Also useful to this review were numerous discussions on these topics in the DOE technical basis report for the TSPA-VA (U.S. Department of Energy, 1998).

The NRC staff found that the proposed methodology for generating external configurations is sufficient to provide reasonable assurance that the analysis has been performed on a comprehensive set of external configurations and that no configuration that could substantially increase the calculated probability or consequence of a criticality event has been excluded. The method is appropriately tied to site and design features and involves a set of geochemical processes that encompasses realistic mechanisms for FM and neutron-absorber fate and transport.

### 3.4.2.2 External Configuration Modeling Approach (Section 4.2.3)

For the portion of the analysis concerned with generation of external configurations, the DOE requested acceptance of the use of a geochemistry-transport code, a geochemistry code used in a mode that simulates transport, or both, to calculate FM accumulation external to the WP (SER Section 3.4.2). Acceptance of specific codes was not requested. Section 3.4.2.1 of this SER discusses the DOE approach to constructing parameter ranges for potentially critical external configurations. The

physical modeling—chiefly geochemical and hydrologic—applied in this effort is discussed separately in the TR. (Note that the present discussion does not include criticality modeling.)

As discussed in TR Section 3.3, the DOE will take a systematic approach to determining the parameters essential to modeling the distribution of FM, neutron absorbers, and other physical and chemical features of external configurations that affect the criticality potential. External modeling will use geochemical models that include relevant geochemical processes and incorporate transport. At the time Revision 0 of the TR was prepared, central to this modeling was an "open system" geochemical transport model that was a manipulation of the zero-dimensional reaction code EQ3/6 (Wolery, 1992) to simulate one-dimensional water chemistry evolution, water-rock interaction, and FM deposition (Section 4.2.3 and Figure 4-7 of the TR). In the response to NRC RAI 4-32, the DOE stated that this model is expected to be replaced by the geochemical transport code PHREEQC (Parkhurst, 1995), supplemented by a modification of EQ3/6 (Letter, November 19, 1999).

Another component of external configuration modeling will be the configuration generator code (Section 3.4.1.2). As for internal configurations, this code will provide bookkeeping for the transport between sites of application of a detailed geochemistry code and, in some situations, provide more rapid calculation where the detailed geochemistry code results can be used to develop heuristic models for the most significant ions for a few solution parameters. Section 4.3.3 of the TR describes how the code will be applied for geochemical tracking along pathways to two external settings, or "pond locations": the invert and the first "pond" external to the drift. For the invert, at each time step, the configuration generator code will accept the inflow from the WP and drift and then calculate the outflow decrement, pH and solubilities, precipitation/dissolution of appropriate solids, and, finally, concentrations in the outflow solution. For the initial path through the rock below the drift, the code will accept the outflow from the invert and then calculate the fracture travel time, matrix travel time, and outflow. The balance between use of the configuration generator code and the geochemistry models discussed in the previous paragraph has not yet been determined.

The DOE provided additional, clarifying information in its responses (Letter, November 19, 1999) to NRC RAI items 3-5, 3-9, 4-29, 4-31, 4-32, 4-37, 4-39, and 4-40, that resolved NRC questions on the external geochemical modeling approach. As noted earlier in this section, the response to RAI 4-32 (among others) states the DOE has changed what specific models it will use. The other RAIs concerned apparent deficiencies in the modeling approach suggesting that the models may not treat comprehensively all factors that can affect contaminant fate and transport. In each case, the DOE stated the relevant processes or mechanisms were, in fact, to be included in modeling as follows:

- RAI 3-5. Adsorption (which is one reason for the expected shift to PHREEQC)

- RAI 3-9. Temperature variation
- RAI 4-29. Kinetics
- RAI 4-31. Colloidal transport
- RAIs 4-37 and 4-39. The range of water geochemical parameters in addition to pH
- RAI 4-40. Matrix-fracture distribution of flow

In each case, the DOE committed to providing language in a future TR revision to clarify that stated processes are merely examples. In many cases, the DOE provided more detailed, illustrative, technical discussions of expected modeling techniques, which made clear that its modeling approach was careful and comprehensive.

Staff assessed the DOE modeling approach for generating external configurations, using the same documents and acceptance criteria listed as the basis for assessment of the methodology in Section 3.4.2.1.

The NRC staff accept the DOE use of a geochemistry-transport code, a geochemistry code used in a mode that simulates transport, or both, to calculate FM accumulation external to the WP. Such codes, when properly validated and applied, can provide reasonable estimates of contaminant (e.g., FM and neutron absorbers) disposition. These codes can also allow calculation of bounding cases that can support an argument of reasonable assurance that no configuration that could substantially increase the calculated probability or consequence of a criticality event has been excluded. Properly applied geochemical codes (in conjunction with natural analog data) provide a bridge between laboratory or field-test data and long-term prediction and have been applied extensively in other components of the repository program (e.g., U.S. Department of Energy, 1998). This approval is contingent on a satisfactory revision, of the TR, that addresses the RAIs discussed in this section.

### 3.4.2.3 External Configuration Validation Approach

The DOE has requested acceptance of the validation approach for the external accumulation modeling methodology and the configuration generator code. In Enclosure 2 of the RAI response, the DOE stated that accumulation modeling validation will include "... comparison between codes (both EQ3/6 and PHREEQC), comparison with experimental data, and comparison with natural analogs." Acceptance was not requested for bounding cases, nor for the results of any validation exercises. In addition, configuration generator code validation was described as involving hand calculations to check species tracking.

Because the DOE is not requesting acceptance of specific codes, this review will view the validation process generically. One component of the process is comparison among different codes; the examples presented by the DOE being between PHREEQC and either the linked EQ6 runs (Section 4.2.3 of the TR) or a modification of EQ6 that incorporates a Lagrangian transport model (response to RAI 4-33). Code comparison

does not provide direct evaluation against actual data, but may provide an extra measure of confidence beyond validation. The second component of validation mentioned in the response to RAI 4-33 is comparison "... against more complex analytical solutions for reactive transport." This process is not described further, and so cannot be evaluated. The third validation component is "... against experimental data (both laboratory and natural analogs)." The RAI 4-33 response lists some examples of such validation for reactive transport codes with reference to the applicability of parameters such as Peclet and Damköhler numbers, flow rates, and water chemistry. Similarly, the response discusses criteria for selecting appropriate laboratory data for validation purposes, and cites ongoing studies that may prove useful for benchmarking. The DOE committed to discuss codes not covered in the TR (PHREEQC and Lagrangian modified EQ3/6) in the revised TR.

Laboratory and natural analog data validation cases for EQ3/6 are depicted in Tables 4-2 and 4-3 of the TR; because specific codes are not under consideration for acceptance in this SER, these tables were reviewed only as examples. NRC RAI 4-33 raised topics of applicability and bounding with regard to these validation cases; the conditions under which EQ3/6 has been validated, as reported in the TR, are not analogous to either the modeled in-drift conditions that may include WP corrosion products or to the external conditions of low-temperature interaction between drift effluent and tuff fractures. The DOE responses discussed in the preceding paragraph were targeted to those critiques. The DOE acknowledged, in TR Section 4.2.4.2, that reaction rates are highly uncertain and not constrained by the validation exercises.

Beyond validation, another means for increasing confidence that the modeling of external accumulations has not resulted in the exclusion of configurations that would increase the probability or consequence of criticality is sensitivity studies. For example, a poorly constrained geochemical parameter, such as a solubility product that controls the quantity of an FM in a certain configuration class, may be varied from one run to another to ensure that model results do not change significantly in response to changes in the parameter value. In its response to RAI 4-34, the DOE described an example sensitivity analysis that, while directed to in-package chemistry, also serves to illustrate external cases. The DOE, however, does not explicitly state that it will perform such analyses for the criticality analysis supporting the LA.

For validation of the configuration generator code, the DOE proposes to use hand calculations. This approach is based on the fact that the configuration generator code is a bookkeeping tool that does not simulate physical or chemical processes—it merely serves to track the results of such simulations.

Staff assessed the external configuration model validation approach with reference to the following IRSR acceptance criteria listed in this SER: ENFE items 12 and 13 (Section 2.3.3) and RT item 4 (Section 2.3.4).

Staff approve the DOE validation process for external accumulation modeling as exemplified by Table 4-2 and Table 4-3 of the TR and as summarized in the response to RAI 4-33 (Letter, November 19, 1999). This acceptance is contingent on:

- TR revisions regarding code descriptions promised in the response to RAI 4-33;
- More detailed description in the revised TR of the geochemical code validation process, perhaps modeled on the discussion in the RAI 4-33 response and including additional details on the comparison against more complex analytical solutions for reactive transport;
- The DOE's use, in the criticality analysis, of the principles of selection of appropriate and bounding laboratory and natural analog data as discussed in that response, in recognition of the insufficiency of the lists in Tables 4-2 and 4-3 of the TR;
- The DOE's use, in the criticality analysis, of uncertainty and sensitivity analyses, providing reasonable assurance that, as a result of uncertainties in parameter values, modeling has not excluded configurations that would increase the probability or consequence of criticality.

Additionally, the proposed methodology for validation of the configuration generator code by comparing the results of the code with appropriate hand calculations is acceptable. In the LA, the DOE will have to provide verification that these computer codes are being implemented correctly and demonstrate that the use of these computer codes does not underestimate the probability of a criticality event for all WFs disposed in the repository.

## 3.5 Criticality Evaluation ( Sections 1.2, 3.4, 3.5, 4.1)

This section of SER provides the staff evaluation of the DOE's proposed methodology for calculating the $k_{eff}$ values for internal and external configurations. With regard to criticality evaluation, the DOE has made the following approval request in Enclosure 2 of the responses to the RAI:

> *DOE requests acceptance of the criticality evaluation process discussed in Section 3.4. This process is illustrated in Figure 3-3 and revised in Attachment C of Enclosure 1. Acceptance of this item is requested in Section 1.2, Part D of the Topical Report. Review and acceptance of specific computer codes and isotope sets are not requested. We are not seeking acceptance of any specific application of the process and recognize that we will need to demonstrate the specific applications to support licensing.*

In this section, the staff also considered the DOE's approach for developing a regression equation for $k_{eff}$ as a function of a set of configuration parameters as proposed under Section 3.5 of the TR. In this regard, the DOE has stated that:

> *DOE requests acceptance of the following aspects of the probability estimation method: (1) Development and use of a table of $k_{eff}$ for the range of possible configuration parameters to construct a regression for $k_{eff}$ as a function of these parameters or for direct table lookup and interpolation (Section 3.5, page 3-21 and modification of this paragraph given in the response to RAI 3-16)....*

The following sections provide a summary of the DOE's proposed methodology, modeling, and validation approaches and the staff evaluation of the above requests.

### 3.5.1 Methodology (Section 3.4)

Section 3.4 of TR provides a description of the methodology for calculating the $k_{eff}$ for a given internal or external configuration. The DOE's proposed methodology for determining the $k_{eff}$s for both internal and external configurations is depicted in Figure 3.5.1-1 in this SER. The following sections provide a summary of the DOE's proposed methodology for material composition, criticality evaluations, the regression analysis, and the staff evaluation of these three areas.

#### 3.5.1.1 Material Composition (Sections 3.4.2, 3.4.3)

As shown in Figure 3.5.1-1, the results from degradation analyses -- which would include results from corrosion, geochemistry, and configuration generation models, along with some isotopic inventories, at a given time-step -- will be provided as the material and geometry input for $k_{eff}$ calculations. In its methodology, the DOE is proposing to take "... credit for the reduced reactivity associated with the net depletion of fissionable isotopes and the creation of neutron-absorbing isotopes during the period since nuclear fuel was first inserted into a commercial reactor." This is referred to as burnup credit. The DOE had requested approval of a specific set of radionuclides for the purpose of criticality analyses. However, as indicated in the DOE's revised requests, through the response to RAI 3-13 and Enclosure 2, DOE is not seeking approval of a set of specific isotopes at this time. It is requesting approval for the use of the burnup credit concept in disposal criticality. The initial isotopic inventory (i.e., isotopic inventory at time of emplacement in a repository) will be established through validated isotopic models that simulate reactor operating history and the impact on the isotopic inventory for commercial and naval SNF. For other waste forms, such as DOE-owned SNF, the DOE is not seeking credit for the previous operating history. The changes in isotopic inventory from decay and degradation processes will be taken into consideration.

The staff evaluated the DOE's proposed methodology with respect to both the internal and external WP material composition, for the initial and postclosure period. The staff

agrees that the configurations resulting from corrosion, geochemistry, and configuration generation models can be used if the validity and accuracy of these models can be established, as discussed in Section 3.4. With respect to the isotopic inventory assumption, the staff accepts the approach of taking into account the reduced reactivity of the SNF due to its irradiation in reactor cores. The extent to which DOE can take credit for the burnup of fuel depends on the level of information to be provided in the LA with respect to predicting the isotopic inventory and $k_{eff}$ accurately. In addition, the irradiation history must be appropriately modeled (e.g., local conditions in the reactor core, presence of neutron-absorbing materials, axial and radial non-uniform burnup distribution, etc.), identifying, quantifying, and including all the uncertainties through model validation as discussed in Section 3.5.1.3. The staff does not approve inclusion of any reactivity-reducing isotopes in the criticality analyses at this point because of the absence of validation reports.

With respect to postclosure isotopic inventory in configurations with intact spent fuel assemblies, the DOE is proposing to take into account the changes in isotopic inventory caused only by decay. The staff believes that the DOE must consider changes in the isotopic inventory, especially the fission products with high solubility, because of the presence of cracks and pin-hole leaks in the cladding if the DOE is planning to take credit for cladding.

## Open Item

- The DOE must include the effects of radionuclide migration from intact fuel assemblies though pin-holes and cracks in the cladding.

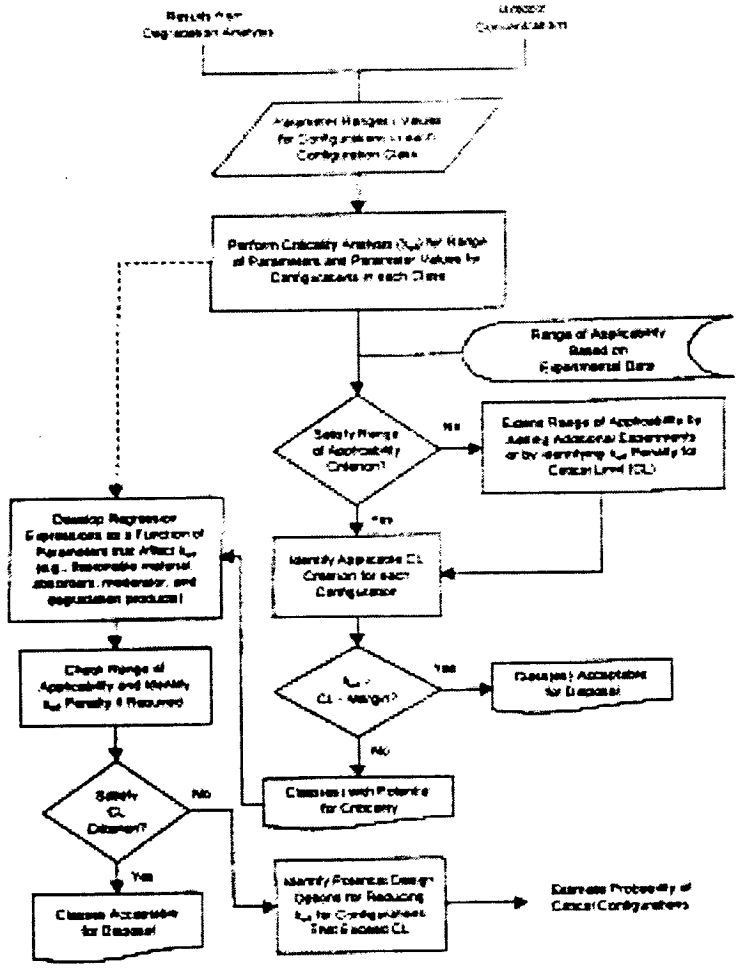


**Figure 3.5.1-1 Criticality analysis methodology**

### 3.5.1.2 $k_{eff}$ Evaluation (Section 3.4.4)

The results from degradation and isotopic analyses and Monte Carlo-based computer codes will be used to determine the $k_{eff}$ for a range of both internal and external configurations. In its response to RAI 1-8, the DOE indicates that the $k_{eff}$ evaluation for the full range means evaluation using the "... bounding values for certain key parameters and the range of values for other parameters." The $k_{eff}$ value for a specific configuration shall be compared with the corresponding CL. The CLs shall be established as described in Section 3.2. In addition, the CLs shall be modified based on the range of applicability of benchmark experiments. As indicated in Figure 3.5.1-1, the modification of CL is in terms of extending its range of applicability through adding additional data points from critical experiments or taking a penalty for the lack of critical experiment. The range of applicability shall be discussed further in Section 3.5.3. If the $k_{eff}$ for any of the bounding configurations exceeds the corresponding CL-margin value, the DOE has proposed to develop a regression equation, for $k_{eff,}$ for that configuration class which is a function of a specified set of parameters (e.g., fissionable material, absorbers, moderator, and degradation products). A comparison of $k_{eff}$ values with the corresponding CL values, modified for the range of applicability, will be performed to identify specifically those configurations with $k_{eff}$ values exceeding the corresponding CL values. Depending on the probability of the configurations with $k_{eff}$ values exceeding the CL values, design modifications or consequence analyses will be performed.

In its examination of the DOE's proposed methodology for performing $k_{eff}$ analysis as depicted in Figure 3.5.1-1, the staff finds the initial parametric $k_{eff}$ analysis to be the crucial step. If the DOE is relying on this parametric $k_{eff}$ analysis to determine which configuration classes have the potential for criticality, the analysis must not be too coarse, which would overlook configuration classes with configurations having potential for criticality. Therefore, the staff acceptance of that portion of methodology related to the initial criticality evaluation for the range of parameters and parameter values of the configurations in each configuration class, depicted in Figure 3.5.1-1, is contingent on the DOE performing fine parametric criticality evaluations.

Furthermore, the criticality margin depicted in Figure 3-3 of the TR should also be used when comparing $k_{eff}$ values from regression analyses with CL values. In addition, the DOE must present an approach for developing the criticality margin.

The range of applicability portion of Figure 3.5.1-1 is evaluated in the Validation Section, and the regression model portion is discussed in the next section.

**<u>Open Items</u>**

- The DOE must include a criticality margin when comparing $k_{eff}$ values from regression analyses to CL values.

- The DOE must present an approach for developing the criticality margin.

### 3.5.1.3 Regression Analysis (Section 3.5)

The proposed methodology is for developing a regression equation, from a set of $k_{eff}$ values, which will be used for determining $k_{eff}$ for a range of internal or external configurations, within a configuration class.

The DOE has proposed first to identify those configuration parameters that impact the $k_{eff}$ values. The DOE has identified a partial list of such parameter types:

1. WF isotopic (based on burnup, enrichment, and time since discharge for commercial SNF);

2. Parameters characterizing the amount of fissile WF;

3. Parameters characterizing the amount and geometry of fissionable material released by the degradation of the fissile WF and remaining in the WP;

4. Parameters characterizing the amount of neutron-absorber material remaining in its intact carrier;

5. Parameters characterizing the amount and geometry of neutron absorber released by the degradation of its carrier and remaining in the WP;

6. The amount of moderator (principally water, but also including the evaluation of silica where appropriate, particularly for external configurations). For potential fast criticalities, the amount of moderator needed would be very low;

7. The amount and distribution of moderator-displacing material (e.g., iron oxide);

8. The amount of neutron reflector material surrounding the fissionable material.

Once all the configuration parameters are determined, a look-up table or a linear regression equation, which provides the values of $k_{eff}$ as a function of the configuration parameters, is developed. This equation or look-up table will be used to determine further the specific configuration in a configuration class that showed the $k_{eff}$ value to be higher than the corresponding CL, using the bounding configuration parameter values. In addition, this regression equation or look-up table will be used in calculating the probability of criticality over a period of the postclosure.

The staff considers the regression or look-up table to be a simplified or abstracted model. Therefore, the applicable acceptance criteria from the five criteria listed under TSPAI, with regard to model abstraction, were used as the basis of the staff evaluation for the look-up table or $k_{eff}$ regression equation.

The DOE must identify all the parameters that would affect the $k_{eff}$ values, and incorporate them in the look-up table or regression equation. Furthermore, the limitation of the regression or look-up table model must be examined over the expected range of configuration parameter values. The data points used in developing the regression equation or the look-up table must be finely partitioned. Therefore, the staff agrees with the approach recommended by the DOE, provided:

- All configuration and waste form parameters affecting $k_{eff}$ are identified; and
- In the case of the look-up table, the interpolation must be within a small interval.

### 3.5.2 Modeling Approach (Section 4.1)

The following sections provide a summary, technical basis, and staff evaluation of the DOE's proposed modeling approach, with respect to isotopic, criticality, and $k_{eff}$ regression analyses. Since no validation reports for any of the specific codes mentioned in TR are available at this time, no evaluation or approval of any specific codes can be provided in this SER. The staff evaluation regards the modeling approach used in the neutronics codes. In the "Validation Section", the staff also evaluated the approach proposed by the DOE for the neutronics model validation rather than evaluation or approval of the validation results for any specific range of applicability.

The following sections are related to the DOE's request stated in Section 3.5. Specifically, the staff evaluation of the DOE's proposed modeling approaches for the items in Figure 3.5.1-1 related to isotopic, criticality, and regression analysis are provided in this section.

#### 3.5.2.1 Isotopic Modeling Approach (Sections 4.1.1, 4.1.1.1, 4.1.1.2)

Part of the DOE's request, stated in Section 3.5, is the use of a model to predict the isotopic inventory, in the WF at the time of emplacement, and at any time during the postclosure. As stated by the DOE in the request, "review and acceptance of specific computer codes and isotope sets are not requested." Therefore, the staff evaluation of the DOE's proposed isotopic modeling is with regard to the approach for predicting isotopic inventory instead of evaluation of specific computer codes.

The DOE is proposing to use the SAS2H module of SCALE 4.3, with the 44-energy group cross-section library (ORNL 1995) to predict both the initial isotopic content of SNF at the time of emplacement, and during the postclosure period. SAS2H is the control module for the analytical sequence. The functional modules invoked by SAS2H are BONAMI, NITAWL-II, XSDRNPM, COUPLE, and ORIGEN-S. BONAMI and NITAWL-II perform problem-dependent resonance weighting of neutron cross-sections. XSDRNPM is a one-dimensional, discrete-ordinates code that produces a weighted cross-section library and spectral data. COUPLE uses these data to update an

ORIGEN-S data library for selected nuclides. ORIGEN-S is a point-depletion and decay code that calculates the time-dependent isotopic concentrations using a matrix, exponential-expansion technique. ORIGEN-S computes the isotopic concentrations (actinides and fission products) for all specified operating histories. ORIGEN-S uses a point-depletion model. Therefore, spacial effects are not specifically modeled.

ORIGEN-S is also used in calculating radioactive decay and the buildup of daughter isotopes after the fuel has been discharged from reactor cores. Therefore, the DOE is also proposing to use ORIGEN-S to calculate postclosure isotopic concentrations. No other changes, other than decay in the isotopic inventory, is assumed, during the postclosure period, in an intact SNF assembly.

The staff evaluated the modeling approach used in the functional modules of SAS2H in SCALE- 4.3 and its appropriateness for predicting the initial and postclosure isotopic inventory to be used for criticality analysis.

With regard to using a one-dimensional type model, the staff has concerns about the approach not taking into account the fuel geometry and spacial effect that would impact the neutron spectrum and consequently the isotopic inventory. These effects become more pronounced in Boiling Water Reactor (BWR) fuel assemblies with much more variation in the neutron spectrum, across the radial and axial regions, because of varying enrichment and absorbers. The DOE has not presented any arguments on how a point-depletion approach can appropriately estimate the isotopic inventory across fuel assemblies with varying neutron spectra.

In addition, the modeling approach must include, at the minimum, the effect of changes in the following parameters on the isotopic inventory:

- Dissolved boron in reactor core;
- moderator density;
- fuel pellet temperature;
- presence of burnable absorbers, power shaping, and control rods;
- axial and radial neutron leakage in reactor core resulting in axial and radial non-uniform burnup; and
- void coefficient and any other features specific to BWR irradiation.

As indicated in the "Methodology" Section, in addition to the decay for the postclosure period, the effects of isotope leaching from the fuel rods -- especially some nuclides such as technetium-99 with high solubility -- shall be quantified as a function of time, in assuming a selected set of isotopes present in the fuel assemblies.

**Open Item**

- The DOE must demonstrate the adequacy of using one-dimensional calculations to capture three-dimensional neutron spectrum effect in their point-depletion

calculation or use two/three dimensional calculations for determining the neutron spectra during the depletion cycles to be used in the depletion analyses.

### 3.5.2.2 $k_{eff}$ Evaluation Modeling Approach (Section 4.1.2)

As stated in Section 3.5, the DOE has requested approval of the use of a criticality model to predict the $k_{eff}$ values of the waste package with a set of isotopic inventory calculated by the isotopic model at the time of emplacement and at any time during the postclosure (Figure 3.5.1-1). As indicated by the DOE in its request, review and acceptance of specific computer codes are not requested. Therefore, the staff evaluation of the DOE's proposed criticality modeling is with regard to the approach for predicting $k_{eff}$ values instead of evaluation of specific computer codes. Furthermore, no specific set of cross-section data is evaluated or approved by the staff at this time. Instead the types of cross-section data were evaluated by the staff.

DOE is proposing to use the MCNP 4B (LANL, 1995) computer code with a set of cross-section data to calculate the $k_{eff}$ values for different WP configuration classes. MCNP is a general-purpose Monte Carlo N-Particle code that can be used for neutron transport, and has the capability to calculate $k_{eff}$ for systems containing fissionable material. The Monte Carlo approach simulates the behavior of individual particles within a system. The Monte Carlo method, as applied to neutrons in an MCNP criticality calculation, is based on following a number of individual neutrons through their transport, including interactions such as scattering, fission, absorption, and leakage. The average behavior of a sample set of neutrons is used to estimate the average behavior of the system with regard to the number of neutrons in successive generations (i.e., effective neutron multiplication factor, $k_{eff}$).

The DOE is proposing to use a cross-section library that has been selected based on the following criteria (CRWM M&O 1998m):

- Evaluated Nuclear Data File/B-V (ENDF/B-V) based cross section libraries were selected for use when available, with the exceptions of H-2, B-11, Zr (natural), Ag-107, Ag-109, Eu-151, and Eu-153;

- either ENDF/B-VI, T-2, or Lawrence Livermore National Laboratory (LLNL)-based cross-section libraries were selected for use when ENDF/B-V-based libraries are not available or selected; and

- ENDF/B-VI, T-2, or LLNL-based cross-section libraries were selected based on number of energy points included in the main energy grid, date of evaluation, and availability of certain data.

Note:

ENDF/B-V or ENDF/B-VI are the Evaluated Nuclear Data Files, a U.S. effort coordinated by the National Nuclear Data Center at Brookhaven National Laboratory.

LLNL: This source refers to the evaluated nuclear data libraries compiled by the Nuclear Data Group at Lawrence Livermore National Laboratory.

T-2: This source refers to the nuclear data evaluations performed by the Nuclear Theory and Applications Group T-2 at Los Alamos National Laboratory.

The staff evaluated the DOE's approach with regard to criticality modeling and the selection of a nuclear cross-section library.

The staff agrees with the selection of a Monte Carlo-based-type modeling, such as MCNP4B, for calculating the $k_{eff}$s of different WF in different internal and external configurations. However, the applicability, appropriateness, and validity of modeling these configurations using a Monte Carlo-based code, such as MCNP4B, has to be demonstrated in the LA.

With regard to selection of an appropriate cross-section library, the staff has no objection to the selection criteria used by the DOE in selecting a set of cross-section libraries. However, examination of the reference material indicated that the cross-section data for the majority of the nuclides are at room temperature. If a cross-section set at higher temperatures is needed (e.g., critical benchmarks at elevated temperatures), additional cross section data at the relevant temperatures must be used, or the impact of temperature on cross-sections, for each relevant nuclide, and subsequently on $k_{eff}$, must be quantified.

**Open Item**

- The DOE needs to use the cross-section data corresponding to the temperature for the WP or critical benchmarks.

### 3.5.2.3 Regression Analysis Modeling Approach (Section C3.3)

As stated in Enclosure 2 of the RAI responses, the "DOE requests acceptance of ... Development and use of a table of $k_{eff}$ for the range of possible configurations parameters to construct a regression for $k_{eff}$ as a function of these parameters or for direct table lookup and interpolation (Section 3.5, page 3-21 and modification of this paragraph given in the response to RAI 3-16)." This section describes the modeling approach for this regression or look-up and interpolation technique. Section 3.5.1.3 of this SER discussed the type of configuration parameters that would impact the $k_{eff}$ and that needs to be included in a regression or look-up table, and provided the staff

evaluation of the regression methodology. This section provides the staff evaluation of the type of regression models or look-up table that the DOE has proposed.

The DOE is proposing to use a multivariate regression approach in calculating the $k_{eff}$ values for the range of configurations that exist in a configuration class. For a selective number of configurations, detailed calculations, using a Monte Carlo-based code, such as MCNP4B, are performed, and subsequently regression analysis is used to predict the intermediate $k_{eff}$ values. As an example, the $k_{eff}$ regression equations for a Pressurized Water Reactor (PWR), with partially and fully degraded basket cases, are provided in Appendix C.

$$k_{eff} + 2\sigma = C_0 + C_1 b + C_2 b^2 + C_3 a + C_3 a^2 + C5 \ln(t) + C_6 \ln(t)^2 + C_7 \ln(t)^3 + + C_8 O + C_9 T + C_{10} T^2 + C_{11} T^3 \qquad \text{Eq. 3.3.5.3-1}$$

$$k_{eff} + 2\sigma = C_0 + C_1 \ln(t) + C_2 b + C_3 a + C_4 \ln(t)^2 + C_5 \ln(t)^3 + C_6 b^2 + C_7 b^3 + C_8 a^2 + C_9 a^3 + C_{10} \ln(t) b + C_{11} \ln(t) a + C_{12} O \qquad \text{Eq. 3.3.5.3-2}$$

where:

b=burnup in Giga Watt days (GWd)/metric tonns of uranium (MTU);
a=initial enrichment in weight percent (0.8% - 4.9%);
t=decay time in years for (8000 -250,000);
T=thickness of borated stainless steel remaining in millimeter (0.1 - 10);
O=either volume-percent oxide for the uniform oxide configuration or fuel-rods; rows covered for the settled cases (30 - 40%, or 3 - 4.5 assembly rows).

The effect of various amounts of boron remaining in solution, in addition to those parameters included in Equation 3.5.2.3-1 and 3.5.2.3-2, for 21 partially and fully degraded PWR basket configurations, are taken into account in terms of correction factors.

$$\text{Corrected } k_{eff} = k_{eff} + \Delta k_{eff} = k_{eff} (1 + \Delta k_{eff}/k_{eff})$$

where:

$$\Delta k_{eff}/k_{eff} = C_0 + C_1 \ln(B) + C_2 \ln(B)^2 + C_3 \ln(B)^3 + C_4 T + C_5 O \qquad \text{for partially degraded basket} \qquad \text{Eq. 3.5.2.3-3}$$

$$\Delta k_{eff}/k_{eff} = C_0 + C_1 \ln(B) + C_2 \ln(B)^2 + C_3 \ln(B)^3 \qquad \text{for fully degraded basket} \qquad \text{Eq. 3.5.2.3-4}$$

In responses to RAI 3-16 and C-4, the DOE has indicated that table look-up may be used because of the fact that there are "... enough cases of $k_{eff}$ calculation (over 2000 for the various waste forms and various degradation parameters) ...".

With regard to using the regression-modeling technique, the staff emphasizes those conditions listed in Section 3.5.1.3. Furthermore, the use of the third-order no-interaction model appears to be an assumption that may not be true. There are different assumptions one can make in building a multi-parameter regression model. One of these assumptions is whether the modeler assumes there should be reacting parameter terms, which model the influence of the independent parameters on each other within the multi-parameter regression model. The appropriate models must also be verified through comparison with specific data points calculated using detailed models for the range of expected configurations.

In addition, the staff is not satisfied that equations such as 3.5.2.3-3 and 3.5.2.3-4 can accurately predict the impact of boron in the solution on the $k_{eff}$ of the waste package. Using a correction factor may not simulate the behavior of $k_{eff}$ appropriately. The DOE needs to provide the technical basis for the proposed approach.

Provided the following conditions are met and open items closed, the staff can accept the DOE's proposed multi-parameter regression technique.

The validity of the assumptions and the models is established;

- The $k_{eff}$ values for the selective number of configurations are based on the initial principal isotopic inventory, calculated using the limiting reactor-operating conditions and the appropriate biases and uncertainties, as defined in the isotopic validation section;

The regression is performed based on the full range of parameter values; and

- Validation of the regression approach covering the compliance period is provided.

**Open Items**

- The DOE must include the cross dependency of configuration parameters for $k_{eff}$ regression equations;

- The DOE must provide the technical basis for the correction factors developed for boron remaining in the solution.

### 3.5.3 Validation Approach (Sections 4.1.1.2, 4.1.3)

In this section, the staff evaluation of DOE's requests regarding any parts of the validation approach for neutronics models is presented. The DOE has presented its revised requests in Enclosure 2 of the responses to the RAI. The staff evaluation is presented with respect to validation approaches for isotopic, $k_{eff}$ evaluation, and regression analysis in the following subsections.

### 3.5.3.1 Isotopic Validation Approach (Sections 4.1.1.2, 4.1.3.1.3, 4.1.3.1.4)

In the responses to the RAIs, the DOE has indicated that it is not seeking approval of the method for validation of the isotopic model, but is only requesting acceptance of the method for confirmation of the bounding-applications model. In Enclosure 2 of the RAI responses, the DOE has stated that:

> *DOE requests acceptance of the three requirements presented in Subsection 4.1.3.1.4 of the Topical Report that describe the acceptance criteria for confirmation of the bounding isotopic model used for burnup credit for commercial SNF. Acceptance of this item is requested in Section 1.2, Part K of the Topical Report. We request acceptance of the method for confirmation of the bounding applications model and not of the method for validation of the isotopic model. Further clarification is provided in responses to RAI 4-1 and RAIs 4-4 through 4-6.*

The three requirements that the DOE has listed in Section 4.1.3.1.4, are as follow:

A. *Reactor operating histories and conditions must be selected together with axial burnup profiles such that the isotopic concentrations used to represent commercial SNF assemblies in waste package design shall produce values for $k_{eff}$ that are conservative in comparison to any other expected combination of reactor history, conditions, or profiles.*

B. *These bounding reactor parameters will be used to predict isotopic concentrations that, when compared to best estimate isotopic predictions of the measured radiochemical assay data or the measured radiochemical data itself, must produce values for $k_{eff}$ that are conservative.*

C. *The values for the isotopic concentrations representing commercial SNF must produce conservative values for $k_{eff}$ for all postclosure time periods for which criticality analyses are performed.*

Aside from isotopic-model validation, there is a set of reactor operating parameters that one needs to run a depletion code. Because the spent fuel assemblies have been irradiated in a variety of reactor-core operating conditions, the DOE needs to determine a set of reactor core conditions that when used in a depletion computer code, would result in a bounding isotopic inventory with respect to criticality. The DOE is proposing to use requirement A for determining the values for parameters such as moderator density, fuel temperature, soluble boron, burnable poisons, control rod histories, and burnup profiles. In Section 4.1.3.1.4, the DOE attempts to describe how these bounding parameter values will be determined, using Commercial Reactor Criticals (CRCs), radiochemical assays, and other means.

The staff agrees with requirement A, but finds the explanation of the requirement provided in Section 4.1.3.1.4, which attempts to discuss how the requirement can be met, vague and unacceptable. The staff finds the responses to RAI 4-4 and 4-5 not to the point and confusing. It is not clear whether the DOE is proposing and asking for acceptance of a method on how to determine the bounding parameters, or is leaving the subject to be discussed in the Validation Report.

In its explanation of requirement A in the TR, the DOE states "The second requirement addresses the problem of using integral experiment (CRCs) exclusively for validating the isotopic model and imposes the additional use of radiochemical assay data for commercial SNF." In its response, to RAI 4-5 (c), the DOE states that "DOE is not currently requesting review and acceptance of the method to be used for validation of the isotopic depletion model and establishment of the isotopic code bias. Furthermore, in responses to other RAIs, DOE has indicated that the isotopic validation method will be demonstrated in the validation reports." Attachment A of the RAI responses lists the types of data and analyses that will be used in performing isotopic validation.

The staff is not clear whether requirement B is proposing to establish the biases and uncertainties of the isotopic model, or simply confirm the bounding reactor parameter values. From the responses, it appears that the DOE is not proposing and requesting approval of an isotopic validation method. In addition, Attachment A of the RAI responses does not present a complete approach for establishing isotopic bias and uncertainties. If requirement B is simply established to confirm that the bounding reactor parameter values are indeed bounding, the staff does not see its necessity, once the isotopic model biases, uncertainties, and applicability are established. Therefore, the staff does not see the real purpose of requirement B, and finds no acceptable method for establishing isotopic depletion model bias and the associated uncertainty.

The third requirement addresses changes to the initial isotopic concentration values, as a function of time, for postclosure. The DOE considered only changes to the isotopic inventory in intact spent fuel assemblies in a flooded WP during postclosure caused by decay. Therefore, the DOE is proposing a method in quantifying the uncertainties associated with the half-life and branching fractions. The method consists of performing many depletion calculations for a given fuel burnup, enrichment, and postclosure period. The half-life and branching fractions for each isotope are allowed to vary randomly over their uncertainty ranges. Then, the isotopic concentrations from each set of depletion calculations are used in a criticality calculation to predict the $k_{eff}$ for that set of isotopic concentrations. When a sufficient number of calculations are performed, a distribution of $k_{eff}$ values is obtained about a mean value. The mean value for $k_{eff}$ for each of these sets will also be compared with the $k_{eff}$ value from a single criticality calculation for each set. Based on these comparisons, any potential bias in the $k_{eff}$ associated with this method for evaluating uncertainties is estimated from the difference between the mean $k_{eff}$ values and the $k_{eff}$ values at nominal conditions.

The staff accepts the proposed methodology for quantifying the uncertainty associated with isotopic decay and branching ratios associated with a depletion code. As stated under the Modeling Approach Section, the effect of the gradual loss of isotopic inventory, from intact assemblies, through the fuel rods with cracks and pin holes, must be considered. Therefore, the validation of such analysis must be also considered.

Based on the staff evaluation of the materials presented in the TR and the RAI responses with regard to the isotopic model validation approach, the following are considered open items.

**Open Items**

- The DOE is required to develop an acceptable methodology for establishing bias and uncertainties for the isotopic depletion model.

- The DOE needs to establish the bias and associated uncertainty regarding the analysis or model keeping track of the isotopic inventory loss, through cracks or pin-holes, within intact spent fuel assemblies.

### 3.5.3.2 $k_{eff}$ Validation Approach

With respect to the approach for validating the model that calculates the $k_{eff}$ of an internal or external configuration, the DOE has made the following request.

> *DOE requests acceptance of the criticality model validation process described in Section 4.1.3. Acceptance of this item is requested in Section 1.2, Part G of the Topical Report and discussed further in the response to RAI 1-3. Specifically, DOE requests acceptance that the process presented in Subsection 4.1.3.2 for calculating the criticality limit values and the general approach presented in Subsection 4.1.3.3 for establishing the range of applicability of the critical limit values define the validation process for the criticality model. This validation process will be followed to calculate critical limit values for specific waste forms and waste packages as a function of the degradation conditions. Further clarification is provided in response to RAIs 4-7 through 4-21. We do not seek acceptance of critical limit values, and we recognize that application to specific postclosure repository conditions will need to be demonstrated prior to licensing.*

The DOE has proposed to use laboratory critical experiments (LCEs) and CRCs for estimating the bias and uncertainties that are associated with $k_{eff}$ values calculated by MCNP4B. These biases and uncertainties are used in establishing CLs for a range of configurations and WFs according to the aforementioned formulas. Attachment A of the RAI responses also describes the types of data and analyses that will be used in preparing a validation report for WPs with commercial SNF covering the range of internal configurations. At this point, the DOE has examined a total of 338 LCEs and

45 CRC statepoints to establish the biases and uncertainties which are associated with MCNP4B $k_{eff}$ values.

The 338 LCEs are composed of the following categories with the number of experiments for each category shown in parentheses.

Homogenous Thermal systems (209)
1. Mixed plutonium and natural uranium (34)
2. Plutonium (73)
3. Highly enriched uranium (Uranium-235) (81)
4. Low-enriched uranium (Uranium-235) (15)
5. Highly enriched uranium (Uranium-233) (6)

Homogenous Fast systems (10)
1. Highly enriched uranium (U-233) (10)

Thermal Arrays (51)
1. Highly enriched uranium (Uranium-235) (22)
2. Intermediate-enriched uranium (Uranium-235) (29)

Thermal Lattices (arrays of fuel rods) (68)
1. Low-enriched mixed oxide (13)
2. Low-enriched uranium oxide (Uranium 235) (55)

The 45 CRC statepoints (measured critical conditions at zero power) consist of data from two Babcock & Wilcox- and two Westinghouse-designed PWRs. The CRC statepoints includes annual, 18-month, and 2-year fuel cycles. The initial Uranium-235 enrichment range from 1.93 to 4.17 wt. percent. The soluble boron concentrations range from less than 400 parts per million boron (ppmB) to over 2300 ppmB. Included in the CRC statepoints are cores at beginning-of-life (BOL), with all fresh fuel; beginning-of-cycle (BOC), with a mixture of fresh and burned fuel; and middle-of-cycle (MOC) to end-of-cycle (EOC), with all burned fuel.

The approach for validation of the model for calculating $k_{eff}$, using the LCE and CRC, is mainly discussed in Section 4.1.3.2 and its associated subsections. As stated in Section 4.1.3.2 of the TR, a critical limit is estimated such that a calculated $k_{eff}$ below this limit is subcritical. This definition is described mathematically by Equation 4-1 of the TR.

$$k_s+\Delta k_s \leq CL, \qquad \text{Eq. 3.5.3.2-1}$$

where:

$k_s$= the calculated multiplication factor of a system to be considered subcritical,

$\Delta k_s$= the uncertainty in the value of $K_s$.

The DOE defines CL as,

$$CL = k_C(x) - \Delta k_C(x) - \Delta k_m, \qquad \text{Eq. 3.5.3.2-2}$$

where:

x = a neutronic parameter used for trending,

$k_C(x)$ = the value obtained from a regression of the calculated $k_{eff}$ of the benchmark critical experiments, or the mean value of the data set, if there is no trend,

$\Delta k_C(x)$ = the uncertainty of $k_C$ based on the statistical scatter of the $k_{eff}$ values of the benchmark critical experiments, accounting for the confidence limit, the proportion of the population covered, and the size of the data set,

$\Delta k_m$ = the additional quantified margin that would reduce the CL (because of the effect on $k_{eff}$ associated with the long-term decay of the radionuclides in the WF, and the effect on $k_{eff}$ associated with extending the range of applicability of the CL beyond the experimental database).

Depending on the trend or lack of it in the bias and uncertainties associated with $k_c$, the DOE has defined three types of CLs. As depicted in Fig. 3.5.3.2-1 (Fig. 4-1 in TR), the type of CL is based on the regression results of $k_c$. The three CLs are described in Sections 4.1.3.2.1, 4.1.3.2.2, and 4.1.3.2.3.

The CL determined based on the Lower Uniform Tolerance Band ($CL_{LUTB}$) method is described in Section 4.1.3.2.1. The DOE is proposing to use the method described in Section 4.1.2 of Lichtenwalter, et al. The formula from Lichtenwalter, et al., for $CL_{LUTB}$, can be inferred as follows:

$$CL_{LUTB} = k_C(x) - (C_{\Delta/P}.S_P), \qquad \text{Eq. 3.5.3.2-3}$$

where the terms on the right side of the equation are defined in Pages 157 and 160 in Lichtenwalter, et al. The DOE has stated that a "no positive bias concept" is maintained (i.e., for $k_c$s that exceed unity, 1.0 will be used).

The CL using the Normal Distribution Tolerance Limit ($CL_{NDTL}$) method is for estimating a CL based on a set of benchmark critical experiments for which normally distributed $k_C$s do not exhibit any trends with respect to any parameters. The CL in this case is established as follow:

$$CL_{NDTL} = k_C - k(\Delta, P, df)*S_P, \qquad \text{Eq. 3.5.3.2-4}$$

where:

$k_C$ = the average of the $k_{eff}$ values for the benchmark set with $k_C$=1 when greater than unity,

$k(\Delta, P, df)$ = a multiplier, with $\Delta$ representing the confidence level, P the proportion of the population covered, and df the number of degrees of freedom,

$S_P$ = the pooled standard deviation.

The third type of CL is the CL determined using the Distribution Free Tolerance Limit ($CL_{DFTL}$) approach. This approach is used when the calculated $k_{eff}$ for the critical benchmarks does not exhibit trends for any of the parameters, and does not pass the normality test.

$$CL_{DFTL} = k_{effl} \qquad \text{Eq. 3.5.3.2-5}$$

where $k_{effl}$ is determined by sorting all the $k_{eff}$ values in ascending order (i.e., $k_{eff1}<k_{eff2}<k_{eff3} \ldots <k_{effN}$) and determining the lth smallest $k_{eff}$ that would establish the limit under which with a predetermined confidence level, a predetermined fraction of future $k_{eff}$ values would fall. The "no positive bias" is maintained by substituting all $k_{eff}$ values that are greater than 1.0, with $1.0 - 3^*\sigma$.

The staff evaluation of the DOE's approach in establishing the CLs was performed using Criteria 7 in the "Container Life and Source Term" IRSR, Regulatory Guide 3.71, ANSI/ANS-8.17, and NUREG/CR-6361. The following is the approach prescribed by ANSI/ANS-8.17 which would be acceptable to the staff.

ANSI/ANS-8.17 recommends the following equation for the purpose of establishing an allowable neutron multiplication factor:

$$k_s \leq k_c - \Delta k_s - \Delta k_c - \Delta k_m, \qquad \text{Eq. 3.5.3.2-6}$$

where, per ANSI/ANS-8.17,

$k_s$ = the calculated allowable maximum multiplication factor, $k_{eff}$, of the system being evaluated for all normal or credible abnormal conditions or events.

$k_c$ = the mean $k_{eff}$ that results from the calculation of the benchmark criticality experiments using a particular calculational method. If the calculated $k_{eff}$s for the criticality experiments exhibit a trend with a parameter, then $k_c$ shall be determined by extrapolation on the basis of a best fit to the calculated values. The criticality experiments used as benchmarks in computing $k_c$ should have physical compositions, configurations, and nuclear characteristics (including reflectors) similar to those of the system being evaluated.

$\Delta k_s$ = an allowance for:
(a) statistical or convergence uncertainties, or both, in the computation of $k_s$,

(b) material and fabrication tolerances, and
(c) uncertainties due to limitations in the geometrical or material representations used in the computational method.

$\Delta k_c$ = a margin for uncertainty in $k_c$ which includes allowance for:
(a) uncertainties in the critical experiments,
(b) statistical or convergence uncertainties, or both, in the computation of $k_c$,
(c) uncertainties due to extrapolation of $k_c$ outside the range of experimental data, and
(d) uncertainties due to limitations in the geometrical or material representations used in the computational method.

$\Delta k_m$ = an arbitrary margin to ensure the subcriticality of $k_s$.

Rearranging Equation 3.5.3.2-6 and including the possibility of trends in the biases as a function of multi-parameters would result in the following definition of CL.

$$k_s+\Delta k_s \leq CL,$$

where:

$$CL=k_c(x,y,z,...)-\Delta\, k_c(x,y,z,...)-\Delta k_m, \qquad \text{for } k_c \leq 1.0, \qquad \text{Eq. 3.5.3.2-7}$$

x,y,z,...=a set of neutronic parameters based on which significant trends are detected,

$\Delta k_m$ = criticality margin,

and the remaining terms are similar to the terms defined per ANSI/ANS-8.17, but with the trending terms x, y, and z.

Equation 3.5.3.2-7 indicates that parameters, against which $k_c$ exhibit significant trends, must be included as the independent parameters in calculating CL. There may be more than one parameter. This is different than what the DOE has proposed. The DOE has proposed in Section 4.1.3.2.1 of the TR "The parameter chosen to trend CL is the one that exhibits the best correlation." In other words, the DOE has proposed to use only one parameter, which exhibits the most trend.

Furthermore, when the value for $k_c$ exceeds 1.0, the parameter value will be kept at the value of 1.0. This would prevent taking credit for positive bias.

The additional margin caused by extending the range of applicability, in the absence of experimental data, shall be included as part of $\Delta k_c$, as indicated by the definition of $\Delta k_c$ under item (c).

The biases and uncertainties established for the principal isotopes, representing the isotopic inventory of a spent fuel assembly (i.e., identified in a previous section as one of the open items), and the uncertainties from decay and branching ratios, described in the previous section, shall be included as part of $\Delta k_c$, under item (d), or an adjustment to the isotopic inventory, in calculating $k_s$.

In Response to RAI 4-9, the DOE has indicated that a criticality margin (shown in Figure 3.5.1-1 as $k_{eff}$ <CL-Margin?) will be used in screening the general configuration classes. As stated in Section 3.5.1.2 of this SER, this margin must be also included in the second screening after regression analysis shown in Figure 3.5.1-1. The staff accepts the DOE's inclusion of a margin in the screening step to compensate for other uncertainties not included in CL, provided this margin is also included in the screening of the configurations. This is in addition to the margin when the range of applicability is exceeded.

In a non-trending situation, the staff agrees with the approach presented by the DOE for the statistical uncertainty portion of $\Delta k_c$ using the NDTL and DFTL in Sections 4.1.3.2.2 and 4.1.3.2.3 of the TR. However, an additional margin in the case of extending the applicability of the benchmarks beyond their range must be included in the CLs.

## Open Items

- The DOE must address the types of criticality uncertainties and biases, which is based on ANSI/ANS-8.17, presented by the staff in this SER.
- The DOE must include a multi-parameter approach in its bias-trending analysis.
- The DOE is required to include the isotopic bias and uncertainties as part of $\Delta k_c$ if not included as isotopic correction factors.

## Range of Applicability (Section 4.1.3.3)

As part of the DOE's request for a validation process for the criticality model, the DOE has requested acceptance of:

> *... the general approach presented in Subsection 4.1.3.3 for establishing the range of applicability of the critical limit values....*

In Subsection 4.1.3.3 of the TR, the DOE presents its approach in identifying and extending the range of applicability of the critical benchmarks with respect to the internal configurations. The DOE is proposing to identify the range of applicability using parameters that represent material, geometry, and the neutron spectrum of the benchmark experiments and the WPs. The DOE has suggested using the "Guidelines for Experiment Selection and Areas of Applicability" discussed in NUREG 6361 (179-

180). The specific parameters and their ranges of applicability will be determined and documented in the criticality model validation report.

In the cases where a specific configuration is beyond the range of the applicable critical benchmark, the DOE is proposing to extend the range of applicability by either including new data in the critical benchmark data set, or by using the established trend to extend the range of applicability to include the calculated data, and to add additional margin in establishing CL. In response to RAI 4-22, the DOE has presented several approaches in how to quantify the additional margin if the method of adding margins is selected. However, the DOE has stated in the response that "... the approach to establishing an additional margin when extrapolation is made beyond the range of applicability will depend on the nature of the bias and the applicable experiments used to establish the bias. Thus, the approach is dependent upon the waste form and its configuration, as well as various aspects of the applicable experiments. The specific approach is, therefore, an application issue for which acceptance is not being sought." The DOE has indicated that the methodology and justification of the specific approach in the appropriate model validation report will be documented and submitted to the NRC as part of the LA.

In its evaluation, the staff has used AC 7 in the CLST IRSR, Regulatory Guide 3.71, ANSI/ANS-8.1-1998, and NUREG 6361.

The approach in demonstrating the applicability of the LCEs and CRCs for validating the MCNP4B for the range of configurations expected to exist within the WP must be similar, with some modifications for repository application, to the approach described in NUREG/CR-6361. In a systematic fashion, all the LCEs and CRCs must be correlated to the specific WP configuration class for which the computer codes biases and uncertainties are calculated. A table similar to Table 4.1 and 4.3, of NUREG/CR-6361, shall be constructed which includes the LCE and CRC categories applicable to the different WP configuration classes. The LCEs and CRCs shall be identified for each WP configuration class, with respect to parameters that would fall within the material, geometry, and neutron spectrum. Material, geometry, and neutron spectrum are the three areas within which trends and range of applicability are identified [i.e., area of applicability (AOA)]. The parameters within each AOA will be identified, and for those parameters exhibiting a trend, their range of applicability to the waste package configurations must be established. Some of the parameters that cover these three areas are identified in NUREG 6361.

The approach for validation of the criticality model, with respect to external configuration, is not addressed specifically in this TR. The work scope provided in Attachment A of the responses to the RAI pertains only to internal configurations.

With respect to extending the range of applicability, the staff agrees that either adding new data in the critical benchmark data set or using the established trend, with additional margin, would be appropriate. However, the DOE has not proposed any methodology for quantifying the additional margin. Therefore, the staff requires, where

the extension of the range of applicability is large, that at a minimum, the DOE's proposed method of using the established trend should be:

(a) subjected to a study of the bias and potentially compensating biases associated with individual changes in materials, geometries, or neutron spectra. This will allow changes which can affect the extension to be independently validated. In practice this can be accomplished in a stepwise approach; that is, benchmarking for the validation should be chosen (where possible) such that the selected experiments differ from previous experiments by the addition of one new parameter so the effect of only the new parameter, on the bias can be observed.

(b) supplemented by alternative calculational methods to provide an independent estimate of the bias (or biases) in the extended area (or areas) of applicability.

**Open Items**

- The DOE must present a validation methodology or work scope for external criticality models.

- The DOE should subject the method used for extending the trend to the procedures defined in ANSI/ANS-8.1-1998, C4(a) and C4(b).

### 3.5.3.3 Regression Analysis Validation Approach

The DOE has not presented any specific methodology or approach for validating the regression technique discussed in the TR. However, the staff requires that, in order for the proposed regression or look-up table to be acceptable, the range of its applicability to the configuration classes has to be established and validated.

The regression or look-up table is considered by staff to be a simplified or abstracted model. Therefore, the applicable acceptance criteria from the five criteria listed under the TSPAI, with regard to model abstraction, were used as the basis of the staff evaluation for the look-up table or $k_{eff}$ regression equation.

As the DOE indicated in its response to RAI 3-16(d), the coefficient of some of the parameters in the regression equation, for WFs with a relatively high plutonium concentration, exhibited some inaccuracies. Therefore, it is very important for the DOE to examine the validity of the proposed regression equations or look-up table over the expected range of WP configurations.

With regard to uncertainty and variability acceptance criteria, the DOE must account for the additional uncertainties introduced by the regression approach or the interpolation used in a look-up table. Furthermore, the addition of the uncertainties to the $k_{eff}$

statistical uncertainties and uncertainties in the configuration parameter values must be included.

**Open Items**

- The DOE must verify the regression equation or look-up table for all ranges of configuration and WF parameters affecting $k_{eff}$.
- The DOE is required to include all uncertainties and variabilities introduced by the regression equation or the look-up table.

## 3.6 Probability of Critical Configurations (Sections 3.5, 4.3, 4.3.1, 4.3.2, 4.3.3, and 4.3.4)

In its responses to the RAI, the DOE has requested an approval of the methodology for determining the probability of critical configurations as stated by the DOE in the following:

> *DOE requests acceptance of the following aspects of the probability estimation method: (1) Development and use of a table of $k_{eff}$ for the range of possible configuration parameters to construct a regression for $k_{eff}$ as a function of these parameters or for direct table lookup and interpolation (Section 3.5, page 3-21 and modification of this paragraph given in the response to RAI 3-16); (2) Monte Carlo methodology using random sampling of parameters characterizing configurations and determination of $k_{eff}$ by calculation from the regression expression or table lookup and interpolation as a function of these parameters to obtain a sample of up to 1 million values of $k_{eff}$ to simulate a probability distribution (the new paragraph for the Topical Report, given in the response to RAI 3-16); (3) Incorporation of the WAPDEG-generated probability distribution for time of breach and duration of the "bathtub" as two of the parameters (Section 3.5, page 3-22); and (4) Estimate of criticality risk for TSPA (before 10,000 years and to the time of peak dose) (paragraph to be included at the end of Section 3.7, attached to the response to RAI C-14). Acceptance of this item is requested in Section 1.2, Part E of the Topical Report.*
>
> *DOE requests acceptance of the validation process for the probability calculation and configuration generator models presented in Sections 3.5 and 4.3 of the Topical Report as modified by responses to RAIs 3-16, 3-19, 4-25, 4-36, and 4-37 that will be implemented by the Monte Carlo probability calculation methodology. DOE plans to validate this methodology by comparison with hand calculations of combinations of probabilities of individual events taken from distributions similar to those used for the Monte Carlo selection process. We also request acceptance*

*that the configuration generator models described in Sections 4.3.3 and 4.3.4 can be validated by appropriate hand calculations.*

### 3.6.1 Methodology

The probability of a critical configuration is determined by first identifying the configuration classes that have $k_{eff}$s exceeding the critical limits over portions of their parameter ranges. This screening uses a multivariate regression for $k_{eff}$ as a function of WF burnup, enrichment, and cooling times. These regressions will be developed using a commercial neutron transport code such as MCNP (Oak Ridge National Laboratory, 1997) for representative values of these parameters and will be based on the upper 95$^{th}$ percent confidence level of the regression such that configuration classes are not likely to be improperly screened because of uncertainty in the regression. The development of the regression equation for $k_{eff}$ is evaluated in Section 3.5 of this SER.

The scenario and configuration parameters are assigned probability distributions based on their uncertainty, and the Monte Carlo technique is used to estimate criticality probability. The Monte Carlo technique consists of a random selection of parameter values from the parameter distributions and determination of whether the selected parameter values satisfy the requirements for criticality. This process of selecting parameter values and determining whether the associated configuration yields a criticality event is repeated many times to yield an estimate of the probability of a critical configuration. Correlations among sampled parameters will be accounted for by using appropriate conditional probability distributions for sampling parameters that depend on previously sampled parameters, as indicated in the DOE response to RAI 4-35. The criticality analysis will use the WAPDEG-generated probability distributions for the time of WP breach and duration of the "bathtub" (i.e., the pooling of water within this WP) associated with the most recent TSPA.

The NRC staff reviewed the proposed methodology to determine the probability of occurrence of a criticality event against the acceptance criteria in the IRSRs. The methodology was reviewed to ensure that it will be sufficient to provide reasonable assurance that the DOE has developed a technically defensible, transparent, and traceable method to assign probability values to criticality scenario classes, scenarios, configuration classes, and configurations (CLST AC No. 6) and has adequately addressed the uncertainty in data caused by both temporal and spatial variations in conditions affecting potential nuclear criticality (TSPAI AC No. 14).

The NRC staff found that the proposed methodology, of using the Monte Carlo technique to account for uncertainty in data, to determine the probability of critical configurations, is acceptable. The NRC staff found this technique will allow the DOE to provide reasonable assurance that the probability of postclosure criticality at the repository will not be underestimated. Acceptance is contingent on the DOE incorporating the steps stated in the response to RAI 3-16 if there is a problem using a regression to represent a parameter. The NRC staff found that the use of the

WAPDEG-generated probability distributions, for the time of WP breach and duration of the "bathtub" inside the WP, associated with the most recent TSPA, is acceptable, provided the DOE can demonstrate that no assumptions were made in the WAPDEG modeling that would be conservative for TSPA calculations, but non-conservative for criticality calculations. The acceptability of the methodology to generate the regression equation to determine $k_{eff}$ has been evaluated in Section 3.5 of this SER. The acceptability of the methodology to determine the risk from criticality has been evaluated in Section 3.8 of this SER.

### 3.6.2 Modeling Approach

Because the potential of a criticality event occurring changes through time as the rate of infiltration to the drift changes, WPs fail, and materials within the WP are redistributed, the neutron multiplication factor must be calculated for many time steps to ensure that the criticality potential of a realization has been evaluated properly. The DOE calculation of the probability of occurrence of an internal criticality will consist of the following steps, as illustrated in Figure 4-8a of the DOE TR:

1. Sample from the distribution of infiltration to the drift from the most recent version of the TSPA;

2. Sample from the distribution of failure times determined by the TSPA programs WAPDEG and RIP, from the drip rate sampled in Step 1. WAPDEG is the code that calculates the failure times and conditions for WP degradation; RIP is the executive driver for the DOE's TSPA program. Per response to RAI 4-25, these failure times will be based on corrosion rates determined by testing programs at LLNL and the University of Virginia;

3. Sample the height of WP penetration to determine the water level in the package;

4. Sample the WF characteristics, including enrichment, burnup, and cooling time, and determine whether this fuel has the potential to yield a criticality event by comparing these characteristics to the bonding characteristics needed to achieve criticality (i.e., the critical limit derived previously). The realization is ended if the fuel cannot yield a criticality event inside the WP;

5. Sample the degradation rates of the WF and the internal components of the WP, accounting for correlations as appropriate;

6. Calculate the amounts of neutronically significant material remaining in the WP, using the degradation state of the WF and the internal components as inputs to the configuration generator code or the detailed calculations of a geochemistry code such as EQ3/6 (Wolery, 1992);

7. Test whether the $k_{eff}$ of the configuration analyzed exceeds the critical limit. The realization is ended if $k_{eff}$ exceeds the critical limit;

8. Check whether the ending condition has been reached and if not, increment the time and return to Step 6. The ending condition is typically a time limit or the time at which a hole develops in the bottom of the WP, water is released, and criticality within the WP is no longer possible.

The probability of criticality will be calculated as the number of realizations that produced a critical configuration of FM divided by the total number of realizations. This process will be repeated for a sufficient number of realizations to yield a sufficiently small uncertainty in the probability of criticality. Per response to RAI 3-16, the DOE has indicated that preliminary estimates of the number of realizations required to drive the uncertainty to an acceptably small number are about $10^8$.

Similar to the calculation of the probability of an internal criticality, the calculation of the probability of occurrence of an external criticality will be conducted with a Monte Carlo calculation and will consist of the following steps, as illustrated in Figure 4-8b of the DOE TR:

1. Sample the flow rate, concentration of FMs, and pH of the water flowing out of the WP;

2. Sample the external path leading to an external criticality location, the transport parameters, and the accumulation parameters. Parameters sampled to determine the location of accumulation include the groundwater flow rate, rock porosity, and the fracture density. Parameters sampled to determine the transport and accumulation properties of materials will include adsorption coefficients and will be consistent with the TSPA;

3. Calculate the amounts of fissionable material removed from the flow in the external environment. Geochemical analyses will be used to identify the portions of the external environment that can remove fissionable material from the flow and determine the chemical environment in these locations;

4. Evaluate the $k_{eff}$ of configurations having a significant accumulation of FM. If the $k_{eff}$ of the maximum concentration of FM exceeds the critical limit, it is recorded and a new realization is started.

The probability of criticality will be calculated as the number of realizations that produced a critical configuration of FM divided by the total number of realizations. This process will be repeated for a sufficient number of realizations to yield a sufficiently small uncertainty in the probability of criticality. The ranges and distributions of most of the parameters sampled will be provided by the inputs into and results of the most recent TSPA.

The NRC staff reviewed the proposed modeling approach to determine the probability of occurrence of a criticality event against the acceptance criteria in the IRSRs. The modeling approach was reviewed to ensure that the DOE has developed a technically defensible, transparent, and traceable method to assign probability values to criticality scenario classes, scenarios, configuration classes, and configurations (CLST AC No. 6) and has adequately addressed the uncertainty in data because of both temporal and spatial variations in conditions affecting potential nuclear criticality (TSPAI AC No. 14).

The NRC staff found that the use of the Monte Carlo technique is an acceptable method to determine the probability of critical conditions occurring based on configurations defined by multiple uncertain parameters. The NRC staff consider the use of data from the most recent TSPA in the criticality evaluation an acceptable source of the input data for the probability calculation by NRC staff as long as correlations among parameters are accounted for in the sampling scheme and the ranges from the TSPA are not conservative estimates for the calculation of dose but non-conservative for criticality calculations. For example, the TSPA may use an unrealistically low value for the $K_d$ of plutonium in the unsaturated zone to be conservative. This value for the $K_d$ of plutonium may not be appropriate in the criticality calculations because a higher $K_d$ could lead to a greater amount of accumulation of plutonium and a higher potential for criticality.

### 3.6.3 Validation Approach

DOE proposes to validate the code that incorporates this Monte Carlo methodology using hand calculations and a commercial mathematical equation solver code to verify that the Monte Carlo code is properly sampling from the input parameter distributions and calculating the probability correctly. One example of the type of hand calculation that could be used in this validation process is fixing the value of sampled parameters to ensure that the code reproduces results that can be verified using an equation solver code.

The NRC staff reviewed the proposed approach -- to validate the models that will be used, to determine the probability of occurrence of a criticality event -- against the acceptance criteria in the IRSRs. The validation approach was reviewed to ensure that the DOE has developed a technically defensible, transparent, and traceable method to assigning probability values to criticality scenario classes, scenarios, configuration classes, and configurations (CLST AC No. 6).

The NRC staff found the methodology of using hand calculations and a commercial mathematical equation solver code, to verify that the Monte Carlo code is properly sampling from the input parameter distributions and calculating the probability correctly, acceptable, provided that a sufficient number of these calculations is conducted to demonstrate that the code is performing the calculations properly across the range of the sampled parameters. In the LA, the DOE will have to provide verification that these computer codes are being implemented correctly and demonstrate that the use of these

computer codes does not underestimate the probability of a criticality event for all WFs that will be disposed of in the repository.

## 3.7 Criticality Consequence Analysis (Sections 1.2F, 3.6, 4.4)

As part of the responses to the RAI, the DOE revised request of 1.2F states that:

> *DOE requests acceptance of the following aspects of the consequence evaluation method for a steady-state criticality: (1) Determination of temperature such that the evaporation rate over the waste package pond surface equals the volumetric drip rate into the waste package (Section 4.4.1.1, page 4-45); (2) Use of the drip rate probability distribution as a function of time (which comes primarily from the climate-and mountain-scale hydrology model)(Section 4.4.1.1 as augmented in the response to RAI 3-23); and (3) Determination of radionuclide increment from depletion code (ORIGEN-S) as a function of power, integrated over the duration of the criticality (Section 4.4.1.1 as augmented in the response to RAI 4-51). Acceptance of this item is requested in Section 1.2, Part F of the Topical Report. It should be noted that acceptance of the ORIGEN-S code will not be sought until the corresponding validation report is complete and referenced in the License Application. The use of this code in the Topical Report, the responses to the RAIs, and supporting documentation, is intended only to provide background for understanding of the methodology.*

As indicated by the request, the DOE is seeking approval of only some specific aspects of criticality consequence analysis at this point. The DOE is specifically requesting the review and approval of the methodology, modeling and validation approach for radionuclide buildup from a steady-state criticality configuration. The DOE has not requested staff review of other aspects of criticality consequences, such as effect of heat from steady-state criticality on the material degradation, methodology, modeling, and the validation approach for the consequence of in-package transient criticalities and external criticalities. Therefore, the NRC staff has not reviewed these aspects of disposal criticalities and no approval is granted at this time.

The following sections provide staff evaluation of what the DOE has provided in the TR and the responses to the RAI, regarding only the build-up of radionuclides from an internal, steady-state criticality event. The evaluation is performed with respect to overall methodology, modeling, and the associated validation approach.

### 3.7.1 Methodology (Sections 3.6, 3.6.1)

In Section 3.6 of the TR, the DOE has stated that "... when the $k_{eff}$ of the configuration analyzed exceeds the CL and the probability of occurrence of that configuration **is below**" (changed from "exceed" which was an error in the TR per DOE's comment on the draft SER), "the waste package probability criterion, currently derived in Section 3.5 as approximately $10^{-4}$ per waste package in 10,000 years, a consequence analysis is

performed." The following section provides the DOE's proposed approach for calculating the radionuclide build-up from a steady-state in-package criticality event.

### 3.7.1.1 Steady-State Criticality Consequence Methodology (Section 3.6.2)

In Section 3.6.2 of the TR, the DOE has proposed an approach for examining the consequences of an internal or external steady-state criticality, with respect to only radionuclide build-up.

The proposed approach consists of first estimating the power and duration of a critical configuration. The power level is determined by the reactivity feedback (the influence of temperature on $k_{eff}$, the heat removal, and the rate of replenishment of the moderator). The criticality duration is most strongly determined by environmental parameters -- particularly the drip rate into the WP, for internal criticality -- or percolation rate into the region of accumulation, for external criticality.

The next step is to compute the total burnup for this power level and duration, using a point-depletion analysis to estimate the increment in radionuclide inventory caused by the criticality event.

The staff evaluated the DOE's proposed approach in evaluating the steady-state criticality consequence for internal and external criticality, regarding only an increase in radionuclide inventory. The staff accepts the approach of using the quantity of water dripping on the waste package as the factor determining the power level and consequently the incremental increase in radionuclide inventory. However, the approach for other types of critical configurations that may involve moderators other than water, especially for external criticality, must be considered. In addition, the loss of soluble neutron-absorbing isotopes from the spent fuel matrix, through pin holes and cracks in the cladding, must also be considered for contribution to the steady-state criticality. Furthermore, other types of steady-state criticality consequences, such as the heat impact of criticality on the material corrosion rate, as acknowledged by RAI 3-23, must be also considered.

Furthermore, the staff agrees that the power can be used as an input to a depletion analysis for determining the incremental increase in the isotopic inventory. However, the modeling and the associated validation, to be discussed in the proceeding sections, must be appropriately performed.

## Open Items

- In developing the methodology for steady-state criticality consequences, the DOE must consider other types of moderators, especially with respect to external criticality.

- The DOE must also consider the loss of soluble neutron-absorbing isotopes through pin holes and cracks in the spent fuel cladding, and its effect on steady-state criticality consequence.

- The DOE must also include other types of steady-state criticality consequences, especially with respect to internal criticality, in its consequence analysis approach.

#### 3.7.1.2 Transient Criticality Consequence Methodology (Sections 3.6.1, 3.6.2)

No specific requests have been made in Enclosure 2 of the RAI responses.
No acceptance or approval is granted.

**Open Item**

- The DOE must develop and present a request for approval of a methodology for transient criticality.

### 3.7.2 Modeling Approach (Section 4.4)

Section 4.4 of the TR provides a brief summary of the DOE's proposed approach in modeling the consequences of steady-state and transient criticality configurations. The following sections provide the DOE's proposed modeling approach and the staff evaluation.

#### 3.7.2.1 Steady-State Criticality Consequence Modeling Approach (Sections 4.4.1.1, 4.4.2.1)

As shown in Section 3.7, the DOE has requested NRC acceptance of specific aspects of the modeling approach for steady-state criticality consequence, with respect to an increase in the radionuclide inventory. These aspects are:

> *(1) Determination of temperature such that the evaporation rate over the waste package pond surface equals the volumetric drip rate into the waste package (Section 4.4.1.1, page 4-45);*
>
> *(2) Use of the drip rate probability distribution as a function of time (which comes primarily from the climate-and mountain-scale hydrology model)(Section 4.4.1.1 as augmented in the response to RAI 3-23);*
>
> *(3) Determination of radionuclide increment from depletion code (ORIGEN-S) as a function of power, integrated over the duration of the criticality (Section 4.4.1.1 as augmented in the response to RAI 4-51).*

The DOE has indicated that acceptance of any specific computer codes, such as ORIGEN-S code, is not being sought at this point.

In its approach for modeling the consequences of a steady-state critical configuration, the DOE assumes that a critical configuration is reached when the waste package is flooded with water and slowly loses boron and iron from the package interior. Once a waste package reaches the CL, continued small positive reactivity insertions will cause the power output of the waste package to begin to slowly rise. Subsequently, the temperature will increase and the evaporative water loss will increase. If the power exceeds a certain limit, the rate at which water is consequentially removed from the waste package will exceed the rate of input, and the resulting water-level drop will provide a negative reactivity insertion, driving the WP back toward a subcritical condition. Conversely, if sufficient power is produced, the water level will be maintained, and the exchange process will continue to remove dissolved boron, thus providing a continued source of positive reactivity insertions until the point of equilibrium is achieved. The maximum steady-state power can then be estimated by determining the power required to maintain the bulk WP water temperature, taking into account the heat loss through radiation, conduction, and convection, at the point where water is removed at the same rate that it drips into the WP. Therefore, the total power rate output from a steady-state criticality is estimated by the following equation:

$$q_{total} = (q_{rad} + q_{cond})/2 + q_{water}, \qquad \text{Eq. 3.7.2.1-1}$$

where,

$q_{total}$ = total heat rate produced by the steady-state criticality configuration
$q_{rad}$ = heat rate loss entirely caused by radiation
$q_{cond}$ = heat rate loss entirely caused by conduction
$q_{water}$ = heat rate loss caused by evaporation

Since the exact steady-state critical configuration cannot be predicted with respect to the amount of rubble covering the WP, the DOE has proposed to use the average value between the rate of heat-loss by radiation, versus conduction mode, as indicated by Eq. 3.7.2.1-1. The DOE is proposing to determine the values for the heat-loss terms by the following equations:

$$q_{rad} = [\sigma.A_1.(T_1^4 - T_2^4)]/[\Delta_1^{-1} + (A_1/A_2)(\Delta_2^{-1}-1)] \qquad \text{Eq. 3.7.2.1-2}$$
$$q_{cond} = [2\Delta.k.L.(T_1 - T_2)]/[\ln(d_2/d_1)] \qquad \text{Eq. 3.7.2.1-3}$$
$$q_{water} = [C_p(30^\circ C).V_{drip}.\Delta T]/\Delta(30^\circ C), \qquad \text{Eq. 3.7.2.1-4}$$

where,

$T_1$ = WP surface temperature
$T_2$ = drift wall temperature
$A_1$ = WP surface area
$A_2$ = drift surface area

$\Delta_1$ = emissivity of outer barrier
$\Delta_2$ = emissivity of tuff rock
$\sigma$ = Stephan-Boltzman constant
k = average thermal conductivity of crushed tuff
$d_1$ = WP outer diameter
$d_2$ = drift diameter
$V_{drip}$ = water dripping into the package
$C_p(30°C)$ = specific heat of water at 30°C
$\Delta(30°C)$ = specific gravity of water at 30°C
$\Delta T$ = temperature increase (27.4°C) (i.e., WP initial water temperature -final water temperature at critical condition)

The DOE assumes the temperature increase ($\Delta T$) is the value that is needed to evaporate only the amount of the incoming water for the WP to stay at a steady-state critical condition. With this assumption, the temperature at which the drip rate is equal to the evaporation rate is calculated by:

$$V_{drip} = V_{evap}(T) = [(D(T).P_{atm}.M_1.A.\Delta(T))/(R_0.T.z)] .\ln[(P_{atm}-p(T).RH)/(P_{atm} - p(T))], \qquad \text{Eq. 3.7.2.1-5}$$

where,

$V_{drip}$ = volumetric drip rate
$V_{drip}(T)$ = volumetric evaporation rate
T = temperature
$D(T) = [435.7.T^{3/2}.(M_1^{-1}+M_2^{-1})^{1/2}]/(P_{atm}.(V_1^{1/3}+V_2^{1/3})$
=diffusion coefficient at temperature T
$P_{atm}$ = atmospheric pressure
p(T) = saturation pressure of water at temperature T
$R_0$ = Universal Gas Constant
z = distance from the water surface to the bulk environment
$\Delta(T)$ = specific volume of the water at temperature T
A = surface area of the water in the WP
RH = drift relative humidity
$M_1$ = molecular weight of water
$M_2$ = molecular weight of air
$V_1$ = molecular volume of water
$V_2$ = molecular volume of air.

It is the above relationship, shown by Equation 3.7.2.1-5, for which the DOE has requested acceptance under the first item, with respect to incremental radionuclide build-up as the result of a steady-state critical configuration.

The staff cannot evaluate the accuracy and acceptability of the above equations and the associated approaches (nor has the DOE requested this) because of the unavailability of any validation results regarding the model. However, the staff

considered the underlying assumption associated with Equation 3.7.2.1-5, per DOE's request.

The inherent assumption in the above approach is that the given configuration has a negative reactivity feedback associated with the changes in the moderator temperature. This may be true for a significant number of WPs with degraded internal components and intact assemblies. However, there may be some configurations with large gaps between the intact assemblies, because of degradation of internal components, which may cause a positive reactivity feedback from the changes in the moderator temperature. Another type of configuration that may not be covered by the above approach is the presence of other moderators such as silica. In its response to RAI 3-21, the DOE indicated that no configuration has been found thus far that would exhibit higher reactivity with moderators other than water.

Provided the DOE continues to identify and examine all possible configurations, especially with respect to moderators other than water, the staff agrees that, for a steady-state criticality condition to be sustained, with water as the moderator with positive reactivity coefficient, the amount of incoming water must be equivalent to the quantity of water being removed from the waste package. A decrease in the drip or evaporation rates could disrupt the sustained criticality conditions.

In its second part of the request, the DOE is proposing the "Use of the drip rate probability distribution as a function of time...." As indicated by the DOE, the drip rate comes primarily from the climate model and mountain-scale hydrology model.

The staff acceptance of the drip rate is dependent on the satisfactory resolution of climate change, hydrologic effects of climate change; present-day shallow infiltration; deep percolation; matrix diffusion; and the other related subissues in the pertinent IRSRs. Therefore, the staff acceptance of the DOE's proposed approach with regard to using the drip rate probability distribution as a function of time is pending the complete and satisfactory resolution of the related subissues.

In its third request item, the DOE is requesting the acceptance of the methodology for "Determination of radionuclide increment from depletion code (ORIGEN-S) as a function of power, integrated over the duration of criticality (Section 4.4.1.1 as augmented in the response to RAI 4-51)." In its response to RAI 4-51, the DOE states that "... the power level used to determine the burnup/depletion will be based on a specified probability distribution," as opposed to the averaging method indicated by Equation 3.7.2.1-1.

The staff considers that the approach in determining the power, shown by Eq. 3.7.2.1-1, is not part of the request at this point. As indicated by the response to RAI 4-51, the approach has changed since the submittal of the TR. The staff agrees that a depletion analysis using the power integrated over the duration of criticality should be performed to determine the incremental increase in radionuclide inventory. However, this agreement is contingent on the following conditions:

- An acceptable approach for including uncertainty of power;
- Demonstration of the appropriateness and validity of the depletion code;
- Inclusion of the impact of heat from steady-state criticality on the corrosion rate, as acknowledged by the response to RAI 3-23.

With respect to the steady-state external criticality consequence modeling approach, the DOE states that "The analysis to determine operating temperature and power level for an external steady-state criticality is similar to that described above for internal steady-state criticality, except that the radiation and buoyant heat convection-heat dissipation mechanisms are not available for external criticality, and the evaporation mechanism is much weaker." Since the steady-state external criticality consequence modeling approach is not part of the DOE's revised request list in the RAI responses, the staff considers this an open item.

**Open Item**

- The DOE needs to develop, and present for acceptance, the modeling approach for an external steady-state criticality consequence.

#### 3.7.2.2 Transient Criticality Consequence Modeling Approach (Section 4.4.1.2)

No specific requests have been made in Enclosure 2 of the RAI responses.
No acceptance or approval is granted.

**Open Item**

- The DOE needs to develop and present, for NRC acceptance, the modeling approach for transient criticality consequence.

### 3.7.3 Validation Approach (Sec. 4.4.3)

In Enclosure 2 of the responses to the RAI, the DOE has requested acceptance of only "... validation process for the steady-state consequence model" at this time. The following section describes the validation approach proposed by the DOE and the staff evaluation of the approach.

#### 3.7.3.1 Steady-State Criticality Consequence Validation Approach

As part of the acceptance requested by the DOE in Enclosure 2 of the RAI responses,

> *DOE seeks acceptance of the validation process for the steady-state criticality consequence model, specifically that computer code can be written to perform the numerical integration of power over time and distribution of drip rates, as well as calculating the heat loss according to well-known physics formulae. This*

*process is described in Section 4.4.1.4 of the Topical Report, as modified by the responses to RAIs 3-23 and 4-51. The resulting program will be checked by hand calculation. It is assumed that no validation is required for the use of well-known physics formulae. DOE plans to validate the use of ORIGEN-S to compute the radionuclide increment from steady-state criticality with available data.*

In its discussion of model validation for steady-state criticality in Section 4.4.3.1, the DOE states that "The methodology is validated by conservatism inherent in the methodology, which serves to define a range of configuration parameters to provide an upper bound on the increase in the radionuclide inventory." The DOE points to the following assumptions, used to determine the criticality power level and its duration, as examples for inherent conservatism.

(a) The criticality duration extends over the period of high infiltration rate; and

(b) The maximum credible water infiltration rate is maintained over the wet part of a climatological cycle.

However, in response to RAI 4-51, the DOE appears to have moved away from the conservatism argument, for validation purposes, because of the change in the modeling approach. The DOE is proposing to use a drip rate probability distribution that is based on infiltration rates of a mountain-scale hydrology model, as opposed to some high infiltration rate. This approach would also eliminate the uncertainty associated with power calculation. Therefore, with regard to the validation of the part of the model that would calculate the power, the DOE argues that using well-known physics formulae for heat loss, combined with verification of the power calculation module of the model using some hand calculations, would provide adequate validation. With respect to the radionuclide production model, the DOE indicates that the model will be validated using available data.

The staff believes that most of the computer models that have been developed are based on well-known physics formulae. It is the combination of these well-known formulae which one uses to build a model and perform validation in order to provide assurance that the combination of the formulae can predict the behavior of the system. Hand calculations are simply model verification rather than model validation. The staff realizes that a steady-state criticality within the Yucca Mountain environment, for thousands of years, is not something for which one can find benchmark experiments or construct scale model experiments. The staff believes that natural analogues may be an approach that the DOE can use to validate the power model and the steady-state criticality consequence model as a whole. The natural analogues may be more appropriate for an integral type of validation of the steady-state external criticality consequence model.

With regard to the approach for the depletion model, the staff assumes the validation approach presented by the DOE for the isotopic section under Section 3.5.3.1 of this

SER would also apply to the depletion part of the steady-state criticality consequence analysis. Therefore, the same conditions and open items identified in Section 3.5.3.1 would apply to this section.

The evaluation of SAS2H validation with respect to initial isotopic inventory was addressed under 3.5.1.1. With respect to postclosure, the isotopic models can be validated only with regard to the decay constants. This validation may be performed by using the chemical assays over a time range, and extrapolation can be made using the established trend.

**Open Item**

- The DOE needs to develop a validation approach for the power model for steady-state criticality consequence.

### 3.7.3.2 Transient Criticality Consequence Validation Approach (Section 4.4.3.2)

No specific requests have been made in Enclosure 2 of the RAI responses.
No acceptance or approval is granted.

**Open Item**

- The DOE is required to develop a validation approach for a transient criticality consequence model.

## 3.8 Criticality Risk (Sections 3.7 and 4.5)

No specific requests have been made in Enclosure 2 of the RAI responses. In the March 22, 2000, Technical Exchange and the comments on draft SER, the DOE has indicated that risk calculations will be presented in TSPA Model Document. However, the staff believes that the components of criticality risks, probability and consequences must be provided in the TR. This is needed from a transparency and traceability standpoint. Furthermore, the DOE has indicated that the probabilities and the consequences of all the critical configurations will be calculated irrespective of probabilities. The DOE/TSPA will then use the probabilities and consequences of a subset of critical configurations, selected based on probabilities and consequence screening criteria in the proposed Part 63, to calculate the criticality risk.

The staff agrees that using the models for calculating criticality consequences, in terms of taking the intermediate consequences (e.g., incremental increase in radionuclide source term) and calculating the dose to the critical group, is a TSPA task. Therefore, the methodology, modeling, and validation approach for transport of radionuclide from the source to the critical group belongs in documents such as TSPA Model Document. However, the rest of the risk analyses, especially implementation of Figure 1-1 in the

RAI responses, belong to the TR. The TR should describe how tables with probabilities and consequence numbers will be developed for each critical configuration.

## Open Item

- The DOE should describe the interface between Figure 1-1 of the RAI responses and the TSPA criticality risk analysis.

# 4 CONCLUSIONS

The staff evaluated the risk-informed disposal criticality analysis methodology planned for implementation in the LA by the DOE, as presented in the TR. The evaluation was limited to assessing the proposed overall methodology and the approach in modeling and validation. Any requests related to specific applications, data, computer codes, specific benchmark experiments, specific isotopes, or any examples were considered to be beyond the scope of the evaluation, for the purposes of this SER. The staff considered only those requests that were specifically stated in Enclosure 2 of the responses to the RAI in this revision of the SER.

The staff evaluated the methodology with respect to its application to mainly commercial SNF (PWRs, BWRs, and Mixed Oxide) and to some extent for a range of waste conditions (intact, degraded, and degradation product) and different possible locations (in-package, near-field, and far-field) as requested by the DOE. The staff expects to evaluate the applicability of any parts of the proposed methodology to the DOE-owned SNF, immobilized plutonium, and vitrified HLW glass, when a separate TR for DOE-owned waste is submitted.

The staff evaluated each area of the proposed methodology in a sequential manner and has identified a number of conditions, with respect to each area, where an agreement was reached. These conditions are mainly the factors that the DOE has already acknowledged it has taken into consideration, in the responses to the RAI. The second part of the evaluation results is the list of "Open Items." Those requests or any other technical issues that the staff could not reach a satisfactory conclusion on are listed as "Open Items" in the pertinent sections and in the "Conclusion" section. The open items include some of the main parts of the methodology (e.g., transient criticality consequence analysis, risk analysis, etc.) that the DOE is planning to present in the submittal planned for late fiscal year 2000. Since the staff understands that other aspects of the methodology, such as specific models and their validation results, will be part of the LA, they were not included in the "Open Items" list. The following is the list of all the unresolved items that have been listed throughout this SER.

## Open Items

1) The staff believes that burnups of spent fuel assemblies must be verified through measurements before their loading into the WP for the purpose of burnup credit verification;

2) The consequence criteria for transient and external criticalities are not addressed in the TR. The DOE must specify if it intends to perform full consequence analyses for transient and external criticality events and include them in TSPA or use some type of criteria for the purpose of criticality control design selection;

3) The DOE needs to provide a modeling approach for igneous-activity-induced criticality;

4) The DOE must include the effects of radionuclide migration from an intact fuel assembly though pin-holes and cracks in the cladding;

5) The DOE must include a criticality margin when comparing $k_{eff}$ values from regression analyses to CL values;

6) The DOE must present an approach for developing the criticality margin;

7) The DOE must demonstrate the adequacy of using one-dimensional calculations to capture three-dimensional neutron spectrum effect in their point-depletion calculation or use two/three dimensional calculations for determining the neutron spectra during the depletion cycles to be used in the depletion analyses;

8) The DOE needs to use the cross-section data corresponding to the temperature for the WP or critical benchmarks;

9) The DOE must include the cross-dependency of configuration parameters for $k_{eff}$ regression equations;

10) The DOE must provide the technical basis for the correction factors developed for boron remaining in the solution;

11) The DOE is required to develop an acceptable methodology for establishing bias and uncertainties for the isotopic depletion model;

12) The DOE needs to establish the bias and associated uncertainty regarding the analysis or model, keeping track of the isotopic inventory loss, through cracks or pin-holes, within intact spent fuel assemblies;

13) The DOE should address the types of criticality uncertainties and biases, which is based on ANSI/ANS-8.17, presented by the staff in this SER;

14) The DOE must include a multi-parameter approach in its bias-trending analysis;

15) The DOE is required to include the isotopic bias and uncertainties as part of $\Delta k_c$, if not included as isotopic correction factors;

16) The DOE must present a validation methodology or work scope for external criticality models;

17) The DOE should subject the method used for extending the trend to the procedures defined in ANSI/ANS-8.1-1998, C4(a) and C4(b);

18) The DOE must verify and validate the regression equation or look-up table for all ranges of configuration and WF parameters affecting $k_{eff}$;

19) The DOE is required to include all uncertainties and variabilities introduced by the regression equation or the look-up table;

20) In developing the methodology for steady-state criticality consequences, the DOE must consider other types of moderators, especially with respect to external criticality;

21) The DOE must also consider the loss of soluble neutron-absorbing isotopes through pin holes and cracks in the spent fuel cladding, and its effect on steady-state criticality consequence;

22) The DOE must also include other types of steady-state criticality consequences, especially with respect to internal criticality, in its consequence analysis approach;

23) The DOE needs to develop, and present for acceptance, the modeling approach for an external steady-state criticality consequence;

24) The DOE must develop and present a request for approval of a methodology for transient criticality consequence;

25) The DOE needs to develop and present, for NRC acceptance, the modeling approach for transient criticality consequence;

26) The DOE needs to develop a validation approach for the power model for steady-state criticality consequences;

27) The DOE must develop a validation approach for a transient criticality consequence model; and

28) The DOE should describe the interface between Figure 1-1 of the RAI responses and the TSPA criticality risk analysis.

## REFERENCES


Letter from C. W. Reamer, NRC, to S. Brocoum, DOE, Subject: "Submittal of U.S. Department of Energy Topical Report on Disposal Criticality Analysis Methodology," February 11, 1999.

Letter from C. W. Reamer, NRC, to S. Brocoum, DOE, Subject: "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange Meeting on Disposal Criticality Analysis Methodology," October 28, 1999.

Letter from C. W. Reamer, NRC, to S. Brocoum, DOE, Subject: "U.S. Nuclear Regulatory Commission Request for Additional Information on the U.S. Department of Energy Topical Report on Disposal Criticality Analysis Methodology," August 18, 1999.

Letter from S. Brocoum, DOE, to C. W. Reamer, NRC, Subject: "U.S. Department of Energy (DOE) Responses to U.S. Nuclear Regulatory Commission (NRC) Request for Additional Information on the U.S. Department of Energy Topical Report on Disposal Criticality Analysis Methodology," November 19, 1999.

Letter from J. J. Holonich, NRC to D. E. Shelor, DOE, Subject: "Transmittal of the Topical Report Review Plan," February 28, 1994.

Letter from C. W. Reamer, NRC, to S. Brocoum, DOE, Subject: "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange Meeting on Disposal Criticality Analysis Methodology," September 2, 1999.

Lichtenwalter, J.J., S.M. Bowmans, M.D. Dehart, and C.M. Hopper. "Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages," NUREG/CR–6361, Oak Ridge, TN: Oak Ridge National Laboratory, 1997.

Los Alamos National Laboratory, " A General Monte Carlo N-Particle Transport Code, LA-12625-M, 1997.

Oak Ridge National Laboratory, SCALE 4.3, RSIC Computer Code Collection, CCC-545, TIC 235920, Oak Ridge, Tennessee: Oak Ridge National Laboratory, 1995.

Oak Ridge National Laboratory. "MCNP 4B-Monte Carlo N-Particulate Transport Code System—Version 4B." RSIC Computer Code Collection, CCC-SYS, Oak Ridge, TN: Oak Ridge National Laboratory, 1997.

Parkhurst, D.L., "User's Guide to PHREEQC—A Computer Program for Speciation, Reaction-Path, Advective-Transport, and Inverse Geochemical Calculations," Water-Resources Investigations Report 95-4227, Lakewood, CO: U.S. Geological Survey, 1995.

Rahimi, M., U.S. Nuclear Regulatory Commission, personal communication to A. V. Gil, U.S. Department of Energy, October 22, 1999.

Selection of MCNP Cross-Section Libraries, CRWM M&O 1998m, B00000000-01717-5705-00099, Rev. 00, Las Vegas, Nevada: Author. MOL.19980722.0042.

U. S. Nuclear Regulatory Commission, "Issue Resolution Status Report, Key Technical Issue: Container Life and Source Term," Revision 2, NRC: Washington, DC., 1999b.

U. S. Nuclear Regulatory Commission, "Issue Resolution Status Report, Key Technical Issue: Total System Performance Assessment and Integration," Revision 2, NRC: Washington, DC., 2000.

U.S. Nuclear Regulatory Commission, NUREG/CR–2300, "PRA Procedures Guide," NRC: Washington, DC., 1983.

U.S. Nuclear Regulatory Commission, "Nuclear Criticality Safety Standards for Fuels and Materials Facilities," Regulatory Guide 3.71, NRC: Washington, DC., Office of Nuclear Regulatory Research, 1998.

U.S. Nuclear Regulatory Commission, "Issue Resolution Status Report, Key Technical Issue: Radionuclide Transport," Revision 1, NRC: Washington, DC., 1999d.

U.S. Nuclear Regulatory Commission, "Disposal of High-Level Radioactive Wastes in a Proposed Geological Repository at Yucca Mountain, Nevada: Proposed Rule," NRC: Washington, DC., 1999a.

U.S. Nuclear Regulatory Commission "Criticality Safety for Handling, Storing, and Transporting LWR Fuel at Fuels and Materials Facilities," Regulatory Guide 3.58, Revision 9, NRC: Washington, DC., 1986b.

U.S. Nuclear Regulatory Commission, "Nuclear Criticality Safety in Operations with Fissionable Materials at Fuels and Materials Facilities," Regulatory Guide 3.4, Revision 2, NRC: Washington, DC., 1986a.

U.S. Nuclear Regulatory Commission, "Issue Resolution Status Report, Key Technical Issue: Evolution of the Near-Field Environment," Revision 2, NRC: Washington, DC., 1999c.

U.S. Nuclear Regulatory Commission, "Recommendations for Preparing the Criticality Safety Evaluation of Transportation Packages," NUREG/CR–5661, NRC: Washington, DC., 1997.

U.S. Department of Energy, "Disposal Criticality Analysis Methodology Topical Report, YMP/TR-004Q, Revision 0," Las Vegas, Nevada: Office of Civilian Radioactive Waste Management, November 1998.

U.S. Department of Navy, "Addendum to the Disposal Criticality Analysis Methodology Topical Report, for Naval Spent Nuclear Fuel, Addendum to YMP/TR-004Q, Revision 0," Arlington, Virginia: Naval Nuclear Propulsion Program, October 1999.

U.S. Department of Energy, "Total System Performance Assessment—Viability Assessment (TSPA-VA) Analyses Technical Basis Document." B00000000–01717–4301–00006, Revision 01, Washington, DC: U.S. Department of Energy, 1998.

U.S. Department of Energy, "Yucca Mountain Features, Events, and Processes Database," Revision 00b, Preliminary Version. Washington, DC: U.S. Department of Energy, 1999.

U.S. Department of Energy, "Comments on U.S. Nuclear Regulatory Commission (NRC) Draft Safety Evaluation Report (SER) for Disposal Criticality Analysis Methodology Topical Report, Revision 0," Las Vegas, Nevada: Office of Civilian Radioactive Waste Management, May 2000.

U.S. Environmental Protection Agency, "Environmental Radiation Protection Standards for Yucca Mountain, Nevada," Washington, DC: U.S. Environmental Protection Agency, 1999.

Wastler, S., U.S. Nuclear Regulatory Commission, personal communication to A. V. Gil, U.S. Department of Energy, May 20, 1999.

Wolery, T.J., "EQ3/6, A Software Package for Geochemical Modeling of Aqueous Systems: Package Overview and Installation Guide (Version 7)," UCRL–MA–110662 Part I, Livermore, CA: Lawrence Livermore National Laboratory, 1992.

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