ISSUE RESOLUTION STATUS REPORT

KEY TECHNICAL ISSUE: CONTAINER LIFE AND SOURCE TERM

Division of Waste Management
Office of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission

Revision 0

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ACKNOWLEDGMENTS

This Issue Resolution Status Report (IRSR) was prepared by staff of the Engineering and Material Section, Engineering & Geosciences Branch, Division of Waste Management. The staff acknowledge the supervision provided by Richard Weller (Section Leader) and N. King Stablein (Acting Branch Chief) and the guidance provided by the High-Level Waste (HLW) Review Board. Ms. Edith Barbely provided her enthusiastic secretarial support. Other members of NRC's and CNWRA's Yucca Mountain Team have also provided input at various stages of completion of this status report. The Performance Assessment and Integration Section of the Performance Assessment and HLW Integration Branch reviewed the drafts in making this IRSR consistent with the other current IRSRs.

1.0 INTRODUCTION

One of the primary objectives of the U.S. Nuclear Regulatory Commission's refocused prelicensing program is to direct its activities toward resolving the 10 key technical issues (KTIs) it considers to be most important to repository performance. This approach is summarized in Chapter 1 of the staff's fiscal year (FY) 1996 annual progress report (NRC, 1997). Other chapters address each of the ten KTIs by describing the scope of the issue and subissues, path to resolution, and progress achieved during FY 1996.

Consistent with 10 CFR Part 60 requirements and a 1992 agreement with the U.S. Department of Energy (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 the licensing proceedings. Issue resolution at the staff level during prelicensing is achieved when the staff has no further questions or comments (i.e., open items) at a point in time, regarding how the DOE program is 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 the DOE points of view. Furthermore, pertinent additional information could raise new questions or comments regarding a previously resolved issue.

An important step in the staff's approach to issue resolution is to provide DOE with feedback regarding issue resolution, before the viability assessment (VA). Issue Resolution Status Reports (IRSRs) are the primary mechanism that the staff will use to provide feedback to DOE regarding progress toward resolving the subissues comprising the KTIs. IRSRs include: (i) acceptance criteria and review methods for use in issue resolution and regulatory review; (ii) technical bases for the acceptance criteria and review methods; and (iii) the status of resolution including where the staff currently has no comments or questions, as well as where it does. Additional information is also contained in the staff's periodic progress reports, which summarize the significant technical work toward resolution of all KTIs during the reporting period. Finally, open meetings and technical exchanges with DOE provide opportunities to discuss issue resolution, identify areas of agreement and disagreement, and develop plans to resolve such disagreements.

In addition to providing feedback, the IRSRs will serve as guidance for the staff's review of information in DOE's VA. The staff also plans to use the IRSRs in the future to develop the Standard Review Plan for the repository license application (LA).

Each IRSR contains six sections, including this introduction as Section 1.0. Section 2.0 defines the KTI, all the related subissues, and the scope of the particular subissue or subissues addressed in the IRSR. Section 3.0 discusses the importance of the subissue to repository performance, including: (i) qualitative descriptions; (ii) reference to a total system performance flowdown diagram; (iii) results of available sensitivity analyses; and (iv) relationship to DOE's Repository Safety Strategy for the Yucca Mountain (YM) site (U.S. Department of Energy, 1998). Section 4.0 provides the staff's review methods and acceptance criteria, which indicate the basis for resolution of the subissue and which will be used by the staff in subsequent reviews of DOE submittals. These acceptance criteria are guidance for the staff and, indirectly, for DOE as well. The staff's technical basis for its acceptance criteria is also included to further document the rationale for the staff decisions. Section 5.0 concludes the report with the status of resolution, indicating those items resolved at the staff level and those items remaining open. These open items will be tracked by the staff, and resolution will be documented in future revisions of the IRSR. Finally, Section 6.0 includes a list of pertinent references.

2.0 KEY TECHNICAL ISSUES AND SUBISSUES

2.1 Primary Issue

The primary issue of the KTI on container life and source term (CLST) is the adequacy of engineered barrier system (EBS) design to provide reasonable assurance that containers will be adequately long-lived, and radionuclide releases from the EBS will be adequately controlled; such that container design and packaging of spent nuclear fuel (SF) and high-level waste (HLW) glass will make a significant contribution to overall repository performance. The requirements for the adequacy of container design and waste form (SF and HLW glass inside pour canisters) are currently addressed in Section 60.135. (However, NRC is developing performance-based, site-specific regulations for the YM site consistent with direction in the Energy Policy Act of 1992.) Other engineered barriers (e.g., drip shields) may be incorporated in the EBS design, but until DOE makes design decisions on these alternatives, this IRSR will focus on the containers and waste forms as the primary engineered barriers.

In the present DOE design (TRW Environmental Safety Systems, Inc., 199[illegible]), the container consists of a 10-cm-thick outer layer made of a corrosion-allowance steel, such as A516 Grade 55 (a wrought C-Mn steel), and a 2-cm-thick inner layer made of a corrosion-resistant Ni-base alloy, such as alloys 625, or C-22. Additional barriers, such as a multipurpose canister (made of type 316L stainless steel) for Navy spent fuel and defense HLW may be present, but they are not currently considered in the DOE or NRC performance assessment (PA).

There are several design concepts for SF and HLW glass containers. These concepts include both uncanistered and canistered designs. The canistered SF disposal container for direct disposal will come in two sizes. The large size will hold either 21 pressurized water reactors (PWR) or 40 boiling water reactors (BWR) assemblies. The small size will hold either 12 PWR or 24 BWR assemblies. The HLW disposal container for direct disposal will hold five HLW canisters. The co-disposal container for DOE or Navy SF and HLW disposal canisters will hold five HLW disposal canisters with a DOE or Navy SF disposal canister inserted in the middle of the HLW disposal canisters.

2.2 Subissues

Figure 1 identifies four subissues deemed important to the resolution of this KTI, and Figure 2 shows the relationship of the subissues to the subsystems of the repository performance. The subissues are posed as questions:

(i) What are the effects of corrosion on the lifetime of the containers and the release of radionuclides to the near-field environment?

(ii) What are the effects of materials stability and mechanical failure on the lifetime of the containers and the release of radionuclides to the near-field environment?

(iii) Is SF sufficiently resistant to degradation to contribute to the control of radionuclide releases to the near-field environment?

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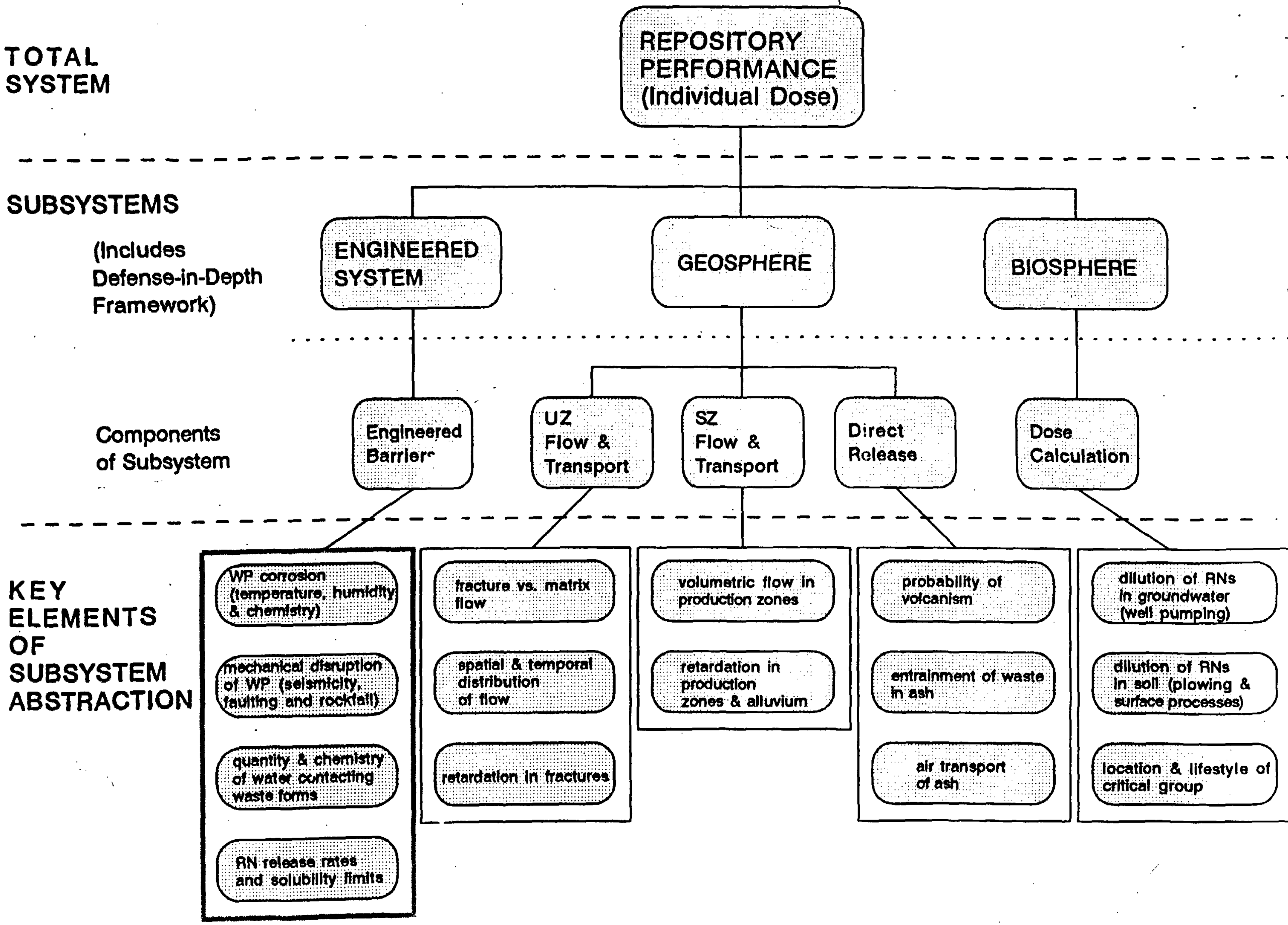

Figure 2. Flowdown diagram for total system performance assessment. The subissues of Container Life and Source Term provide input to the highlighted key elements.

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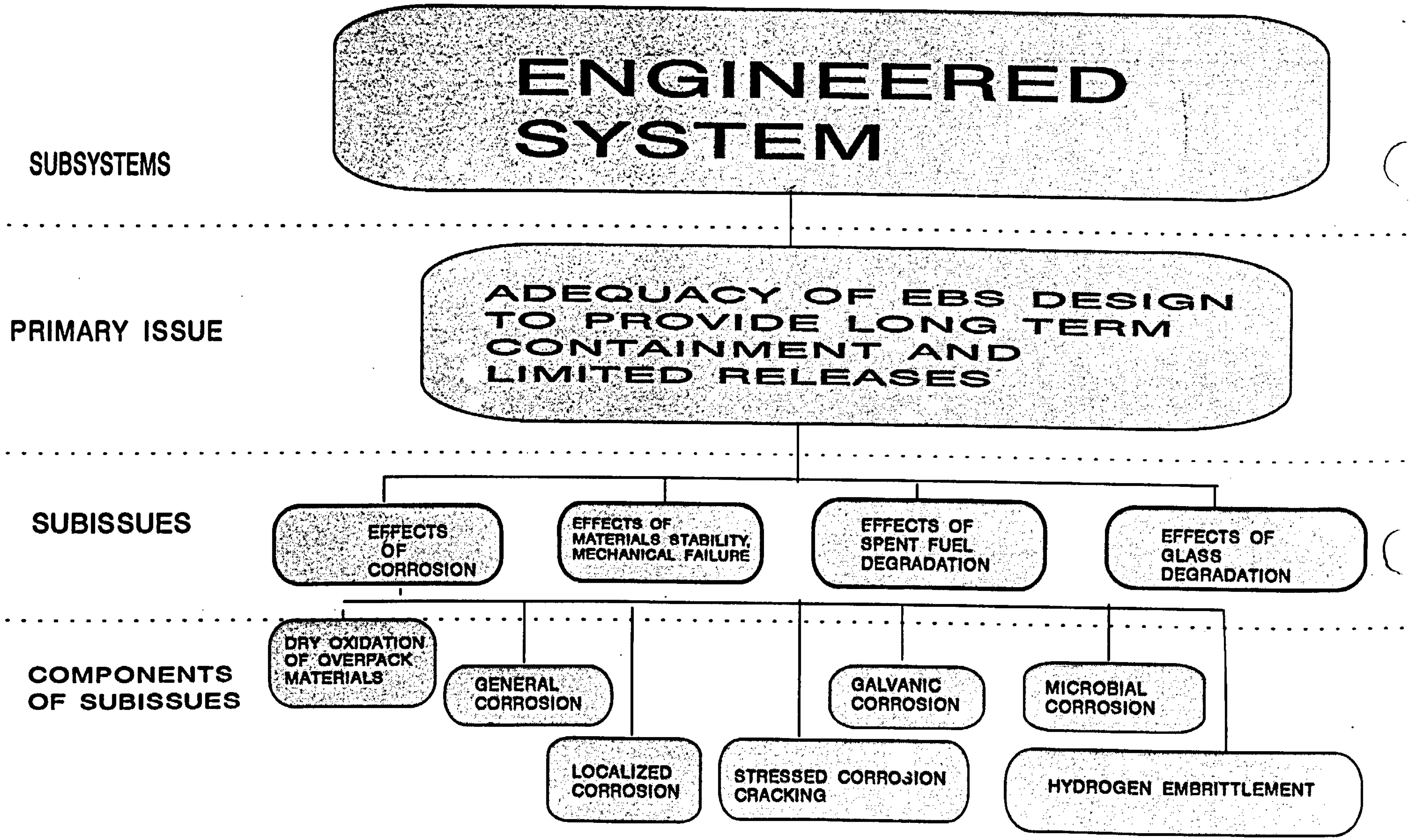

Figure 1. Flow diagram for Container Life and Source Term Subissues

(iv) Is HLW glass sufficiently resistant to degradation to contribute to the control of radionuclide releases to the near-field environment?

Each of the four subissues may, in turn, be addressed in terms of its principal components. Subissue (i) considers container failure by various corrosion modes and consists of seven components: 1) dry air oxidation (of container materials); 2) humid air corrosion and aqueous corrosion, including general corrosion; 3) localized corrosion; 4) stress corrosion cracking; 5) galvanic corrosion; 6) microbial corrosion; and 7) hydrogen embrittlement. Subissue (ii) considers long-term degradation of material properties of containers subjected to elevated temperatures for a prolonged period. The degradation may include redistribution and segregation of chemical species and may result in thermal embrittlement leading to early mechanical failure by brittle failure. Subissue (iii) considers degradation of SF and subsequent radionuclide release from SF, in both dry air and aqueous environments following cladding failure. It involves radionuclide release by matrix dissolution, solubility limit, colloid formation, dry oxidation (of SF), and gaseous diffusion. The degradation of SF may give rise to criticality problems, which initially may occur within the WP, and later on, outside the WP after transport and redeposition of degraded SF elements in the repository environment. Within-package criticality is evaluated as a SF degradation subissue in this IRSR. Extra-package criticality is evaluated as a subissue in the Evolution of the Near-Field Environment IRSR. Finally, Subissue (iv) considers degradation of HLW glass, similar to that of SF. It involves radionuclide release by matrix dissolution, solubility limit, and colloid formation. The staff recognizes that there are additional subissues of interest to this IRSR and other IRSRs in relation to the adequacy of DOE's containers and repository designs for prevention or control of potential criticality events. However, these subissues will not be evaluated in detail until the staff establishes clear postclosure criticality control requirements in the forthcoming YM site specific rule, currently under development.

This version of the IRSR addresses a component of subissue (i) above. It focuses on the significance of dry oxidation of container materials during the dry period of the proposed YM repository. The final version of this IRSR will address all components of all four subissues, and the staff will verify the extent to which DOE has provided adequate technical bases for resolution of each subissue. Furthermore, the staff will confirm that the bases reflect NRC observations, important physical phenomena and processes, consistent assumptions and definitions, consideration of alternative models, bounding approaches, adequate abstraction of process models, appropriate expert judgments, and adequate documentation.

3.0 IMPORTANCE TO REPOSITORY PERFORMANCE

The primary goals of DOE's Repository Safety Strategy (U.S. Department of Energy, 1998) are the near-complete containment of radionuclides within the containers for several thousand years and acceptably low annual doses to the average member of a critical group living near the site. The staff is developing a strategy for assessing the performance of a proposed HLW repository at YM. As currently visualized by the staff, the elements of this strategy that are necessary to demonstrate repository performance are defined as key elements of the subsystem abstraction (KESAs). Figure 2 illustrates the (solid block) KESAs for this KTI. The acceptance criteria, upon which staff review of key elements in DOE's VA and LA will be based, are under development. As noted in Section 2.0 of this report, the subissues related to container lifetime and radionuclide release rates from the EBS are currently considered important factors in the repository performance. For DOE to adequately demonstrate and quantify the consequences that container failure and radionuclide release have on repository performance in its VA and LA, it must consider the effects of the near-field environment on container corrosion, the mechanical disruption of containers, the effects of both quantity and chemistry of the water contacting the waste forms, and the processes that affect solubility limits and radionuclide release rates.

3.1 Relationship of Subissues to DOE's Repository Safety Strategy

The performance of the engineered barriers after emplacement is extremely important in DOE's Repository Safety Strategy for the YM site (U.S. Department of Energy, 1998). In this strategy, DOE has formulated several hypotheses, that if correct, would demonstrate that waste can be isolated at the proposed YM site for long periods of time. These hypotheses state that: (i) heat produced by emplaced waste will reduce relative humidity at the waste package (WP) surface; (ii) corrosion rates are very low at low relative humidity; (iii) double-walled packages will significantly increase containment times due to protection of the inner barrier by the outer barrier; (iv) engineered enhancements can extend the long period of containment of the inner barrier; (v) containment time will be sufficient to prevent oxidation of SF during the thermal period; (vi) the amount of water that contacts waste can be limited; (vii) release rate of soluble radionuclides will be controlled by slow dissolution of the waste form; and (vii) release rate of actinides will be controlled by solubility limits rather than by colloidal stability. The staff needs to evaluate the CLST KTI subissues to determine the merits of DOE's hypotheses and may perform its evaluation using methodologies independent of the ones used by DOE.

3.2 Importance of Subissues to Total Repository System Performance

NRC staff is developing a strategy for assessing the performance of the potential HLW repository at YM. The framework for this strategy is discussed in the "Total System Performance Assessment and Integration" IRSR. As highlighted in Figure 2, the container and waste forms are design features contributing to the four KESAs under engineered barriers. Figure 1 identifies the four subissues considered dominant by NRC staff in determining the adequacy of container design and waste form packaging for long-term containment and limited releases. The container is the primary design element that provides radionuclide containment. After loss of containment, radionuclide release rates from the EBS are limited by waste form characteristics and transport processes through the container and the EBS. The combination of long-lived containers and low degradation rate waste forms can make a significant contribution to the repository system performance. The importance of the CLST subissues to repository performance is discussed in detail below.

3.2.1 Importance to Performance of the Effects of Corrosion

Under anticipated repository conditions, corrosion is expected to be the most dominant failure mechanism for the container (U.S. Nuclear Regulatory Commission, 1997). Loss of containment allows release of radionuclides to the near-field environment. Container failures resulting from disruptive events, are to be considered in other KTI IRSRs.

In recent PA studies (Wilson, et al., 1994; TRW Environmental Safety Systems, Inc., 1995; U.S. Nuclear Regulatory Commission, 1995; Costlier and McGuire, 1996; TRW Environmental Safety Systems, Inc., 1997), container life times have been determined mainly by material corrosion times. Until the container is breached by through-wall corrosive penetration, radionuclide release cannot take place.

3.2.2 Importance to Performance of the Effects of Materials Stability and Mechanical Failure

Staff evaluations (Cragnolino, et al., 1996) indicate that low-alloy steels, such as A387 Grade 22 and A516 Grade 55, may be susceptible to a substantial degradation in toughness as a consequence of long-term aging. This phenomenon, named thermal embrittlement, has been shown to affect tempered low-alloy steels as a result of isothermal heating or slow cooling within the temperature range of 325-575°C. Further, thermal embrittlement may occur at even lower temperatures over longer periods of time. The degradation of mechanical properties can adversely impact container performance and, ultimately, repository system performance.

3.2.3 Importance to Performance of the Effects of SF Degradation

Following container failure, SF will be exposed to the potentially degrading effects of the near-field environment. Potential degradation mechanisms include SF matrix dissolution, dry oxidation, and the formation of colloids and secondary minerals that can result in the mobilization of radionuclides for potential release to the near-field environment (U.S. Nuclear Regulation Commission, 1997; Ahn, 1996a, b; and Manaktala, 1993). The degree to which the SF resists degradation will support the EBS control of radionuclide release to the near-field environment and enhance the overall performance of the repository.

3.2.4 Importance to Performance of the Effects of Glass Degradation

Following container failure, the glass waste form will be exposed to the potentially degrading effects of the near-field environment. Potential degradation mechanisms include glass matrix dissolution and the formation of colloids and secondary minerals that can result in the mobilization of radionuclides for potential release to the near-field environment (Manaktala, 1992). In this regard, the staff recognizes that glass wastes will constitute only about 3 percent of the total radionuclide inventory in the repository. However, if the glass waste form performs poorly in the repository environment, it could conceivably make a significant contribution to the overall radionuclide source term from the EBS. Accordingly, the degree to which the glass waste resists degradation will support the EBS control of radionuclide release to the near-field environment and enhance the overall performance of the repository.

3.3 Consideration of Container Life and Radionuclide Release in Previous PAs

The effect of container life and radionuclide release from the EBS on the performance of the proposed YM repository has been addressed in recent PA studies in which the current conceptual design of the WP for SF and vitrified HLW is considered. These studies include the DOE "Total System Performance Assessment-1995" (TSPA-95) (TRW Environmental Systems, Inc., 1995a); the Electric Power Research Institute (EPRI) "Yucca Mountain Total System Performance Assessment, Phase 3" (Kessler and McGuire, 1996); and NRC and the Center for Nuclear Waste Regulatory Analyses (CNWRA) "Total System Performance Assessment (TPA) Version 3.0 Code: Module Descriptions and User's Guide" (Manteufel, et al., 1997).

U.S. Department of Energy "TSPA-95"

Container life is evaluated in "TSPA-95" (TRW Environmental Systems, Inc., 1995a) using Version 1.0 of the stochastic Waste Package Degradation (WAPDEG) code (Atkins and Lee, 1996). The WP environment in WAPDEG is assumed to be humid air at elevated temperatures for an extended period followed by an aqueous environment, a hypothesis consistent with the DOE Repository Safety Strategy (U.S. Department of Energy, 1998). Dry-air oxidation of container materials is considered to be negligible, and it is not included in the calculations. Humid-air corrosion is distinguished from aqueous corrosion by using two threshold values of relative humidity (RH) at which each corrosion process is initiated. For the carbon steel outer overpack, active general corrosion is assumed to occur in humid air and is modeled using a parametric equation exhibiting a dependence of the corrosion rate on time, RH, and absolute temperature. The parameters were obtained by multiple linear regression analysis of atmospheric corrosion data from tropical, urban, rural, and industrial locations. Aqueous corrosion of the outer overpack is evaluated through a similar approach but using literature data acquired in polluted river water and tropical lake water. In both types of environments, pitting corrosion of carbon steel is modeled by multiplying the uniform corrosion rate by a pitting factor that is assumed to be normally distributed with a mean of 4 and a standard deviation of 1. For the inner overpack material, only pitting corrosion in the aqueous environment is considered, assuming that humid-air corrosion and general corrosion are negligible. The pit growth rate was calculated through an empirical expression following an Arrhenius dependence on temperature. No pit initiation time is considered, and it is assumed that pits nucleate with a uniform distribution on the entire WP surface. In addition, cathodic protection of the inner overpack is evaluated by assuming that pitting corrosion would be delayed until the thickness of the outer metallic barrier is reduced by 75 percent. WAPDEG is a probabilistic code designed to run stochastic simulations in which random values are sampled to represent the parameters in the corrosion models for determining the WP failure time.

Radionuclide release calculations are conducted as part of the Repository Integration Program (RIP), the computer code used for the PA of the repository. WP failure times, along with matrix alteration/dissolution rates calculated by using a parametric equation that depends on environmental factors, are used to calculate the rate at which radionuclides are released, taking into consideration their solubility as a constraint. Finally the rate of mass transfer out of and away from the WPs is computed.

EPRI PA

Another alternative PA of WPs has been presented by EPRI in the "Yucca Mountain Total System Performance Assessment, Phase 3," using the Integrated Multiple Assumptions and Release Calculations (IMARC) Code, as reported by Kessler and McGuire (1996). This is a deterministic code in which an event tree approach is used and the container life is assumed to be governed by a series of Weibull distributions which are dependent on the heat transfer mechanism and the temperature history. Corrosion processes considered in this statistical approach are general corrosion, localized corrosion (pitting and crevice), stress corrosion cracking, degradation due to a metastable microstructure, embrittlement due to hydride formation, and microbially influenced corrosion. Galvanic protection is not considered. The Weibull distributions employ a feature that allows for the possibility that a small fraction of the container may have failed at emplacement, or shortly thereafter, due to manufacturing flaws, construction errors, or emplacement mishandling. The parameters for the distribution in the case of aqueous corrosion processes were obtained through correlations derived from underground corrosion tests in soils.

The source term model is a compartment model in which it is assumed that all waste form surfaces are wetted immediately after the container fails. Advection and diffusion between the following compartments can be modeled: waste form, corrosion products found in the corroded section of the container, gravel backfill below and sometimes above the container, concrete invert (both concrete matrix and fracture), and rock matrix and fractures immediately surrounding the drift. The flux entering the container is assumed to be 5 percent of the wet percolation rate times the horizontal cross sectional area of the container. An opening at the bottom of the container is assumed to be equal in size to the opening at the top, so a flow-through model is used to mobilize waste inside the container. Radionuclides are assumed to be released congruently with the degradation of cladding and dissolution/alteration of the SF matrix, but are constrained by their solubilities.

NRC/CNWRA TPA Code Version 3.1

The engineered barrier system failure (EBSFAIL) module in TPA, Version 3.1 is used to calculate the failure time of the WPs due to various corrosion processes. Below a critical value of relative humidity, which can be selected as an input parameter, only dry-air oxidation takes place. Above this value, humid air corrosion occurs, and at a higher critical value, aqueous corrosion begins. The aqueous environments considered in EBSFAIL are those derived, adopting several simplifications, from coupled thermal-hydrological-chemical calculations. The aqueous corrosion processes for both the outer and inner overpack are governed by the corrosion potential and the critical potential required to initiate localized corrosion. This approach implies the use of well-established electrochemical kinetics equations for calculating the corrosion potential, which depend on environmental variables, such as temperature, oxygen partial pressure, and pH, and experimentally measured values of the critical potentials. The repassivation potential, which, in turn, depends on temperature and chloride concentration, is the critical potential used to define the occurrence of localized corrosion. The initiation time for pitting corrosion, once the corrosion potential exceeds the repassivation potential, is assumed to be negligible, but pit growth rates are calculated by using experimentally determined expressions and parameters. Failure of the WP is defined as penetration of both overpacks by a single pit or by general dissolution. The beneficial effect of galvanic coupling on the inner overpack is assessed through an equation that computes the couple corrosion potential using experimental values from the literature and an efficiency coefficient as an input parameter. A simplified mechanical failure model is included in EBSFAIL to consider the propensity of the outer steel overpack to fracture as a result of thermal embrittlement.

The engineered barrier system release (EBSREL) module in TPA, Version 3.1 calculates the time dependent release of radionuclides after EBSFAIL determines the failure time of the WP. The release calculations are based on the congruent dissolution of the SF, limited by solubility, considering the dissolution rate according to three optional models. One model is based on a parametric equation including environmental factors, the second one on the dissolution in groundwater containing specific species (e.g., Si and Ca) and the third one on the use of any specified value (i.e., from natural analog data). Release from a perforated container can be optionally evaluated through either a bath tub or a flow-through model. Fina''., advective and diffusive transport or radionuclides away from the EBS is computed following their mass balance in the water contacting the WP. Both EBSFAIL and EBSREL are deterministic modules incorporated in the TPA Code.

3.4 Sensitivity Analyses

The results of sensitivity analyses will be provided in future IRSRs. Specifically, the effects of container failure and radionuclide release from the EBS on repository performance will be assessed in terms of sensitivity to individual dose to the average member of a critical group. This effect and the importance of parameter values assigned to physical properties in the analyses are determined by systematically performing sensitivity analyses. Both process-level models and the abstracted models in the PA code can be used to assess the effects of container failure and radionuclide release expected to take place in the repository. Process-level models used by the CLST KTI are detailed models formulated on basic principles that govern container failure and radionuclide release for the range of expected conditions at the repository. Abstracted models within the NRC PA code (Manteufel, et al., 1997) are designed to represent the physical processes by extracting only higher order effects identified in process-level models. The CLST process-level models have been described in the "Engineered Barrier System Performance Assessment Code: EBSPAC Version 1.1, Technical Description and User's Manual" (Mohanty, et al.,1996).

In general, process-level model sensitivity analyses will be provided in future revisions of this IRSR to assess the effects of container failure and radionucide release on repository performance, and corresponding abstracted model sensitivity analyses will be provided in the "Total System Performance Assessment and Integration" KTI IRSR on model abstraction. These studies are currently underway.

The following is a list of the CLST sensitivity analyses to be conducted for the aforementioned four subissues:

3.4.1 Analysis of the Effect of Corrosion Parameters on Container Lifetime

Corrosion parameters to be analyzed may include relative humidity, galvanic corrosion efficiency, critical potential for localized corrosion, passive current density, rate of pit propagation, salt solution concentration, oxygen partial pressure, and oxygen diffusivity.

3.4.2 Analysis of the Effect of Mechanical Failure and Thermal Stability on Container Lifetime

Controlling parameters to be analyzed may include the diffusivity of impurities in the matrix container material, concentration of elemental species, fracture toughness, prior corrosion penetration, such as pit depth, and the effect of manufacturing defects.

3.4.3 Analysis of the Effect of SF Degradation on Radionuclide Release Rates

Parameters to be analyzed may include chemistry of the water contacting the waste form, temperature and surface area of SF, the degree of cladding protection, and criticality.

3.4.4 Analysis of the Effect of HLW Glass Degradation on Radionuclide Release Rates

Parameters to be analyzed may include the radionuclide spectrum and inventory, chemistry of the water contacting the waste form, temperature and surface area of HLW glass.

4.0 ACCEPTANCE CRITERIA AND REVIEW METHOD

The Commission's policy with respect to repository performance is that DOE must be able to demonstrate that the engineered and natural barriers each make a significant contribution to overall repository system performance. In this regard, the CLST primary issue (Section 2.0) relates to the adequacy of the EBS design to comply with the policy stated above. Specifically, the adequacy of the EBS design will depend, in part, on DOE's demonstration that the containers will be sufficiently long-lived and that radionuclide releases will be sufficiently controlled such that the EBS makes a significant contribution to overall repository system performance. DOE must address the four subissues described in Section 2.0, all of which relate directly to processes and events that affect container lifetime and radionuclide release. Resolution of these subissues will also address many of the design criteria for the WP, including the waste form, in the existing rule (Section 60.135). The staff has developed acceptance criteria that, if satisfied, would resolve the CLST subissues, primary issue, and, ultimately, questions related to the adequacy of the EBS design. The acceptance criteria are of two types--general and specific. The general or broader-level acceptance criteria are applicable to all of the CLST subissues and are identified below. These general criteria are supplemented by additional specific acceptance criteria developed fc. each of the four subissues as provided in Sections 4.1 through 4.4.

Acceptance Criteria Applicable to All Four Subissues

(1) The collection and documentation of data, and the development and documentation of analyses, methods, models, and codes were obtained under approved quality assurance and control procedures and standards.

(2) If used, expert elicitations were conducted and documented in accordance with the guidance in NUREG-1563 (Kotra, et al.,1996), or other acceptable approaches.

(3) Sufficient data (field, experimental, and/or natural analog data) are available to adequately define relevant parameters for the models used to evaluate the subissues.

(4) Sensitivity and uncertainty analyses (including consideration of alterative conceptual models) are used to determine whether additional new data are needed to better define ranges of input parameters.

(5) Parameter values, assumed ranges, test data, probability distributions, and bounding assumptions used in the models are technically defensible and reasonably account for known uncertainties.

(6) Mathematical model limitations and uncertainties in modeling are defined and documented.

(7) Primary and alternative modeling approaches consistent with available data and current scientific understanding are investigated, and their results and limitations are appropriately considered in evaluating the subissue.

(8) Model outputs are verified through comparisons to outputs of detailed process models and/or empirical observations.

(9) Model outputs adequately incorporate important design features, physical phenomena, and coupled processes.

4.1 Subissue 1: What are the Effects of Corrosion on the Lifetime of the Containers and the Release of Radionuclides to the Near-Field Environment?

This subissue relates to the adequacy of DOE's consideration of the effects of corrosion on the lifetime of the containers and the release of radionuclides from the EBS to the near-field environment. Resolution of this subissue will be through the application of the generic acceptance criteria specified in Section 4.0 and the specific acceptance criteria identified in Section 4.1.1.

4.1.1 Acceptance Criteria for Subissue 1

(1) DOE has identified and considered critical likely modes for container material degradation, including dry air oxidation, humid air corrosion, and aqueous corrosion processes (i.e., general corrosion, localized corrosion, stress corrosion cracking, galvanic corrosion, microbial corrosion and hydrogen embrittlement).

(2) DOE's numerical corrosion models are adequate representations of expected container performance that are not likely to underestimate the actual performance of the containers in the repository environment.

(3) DOE has considered the compatibility of container materials and container fabrication processes in the performance of their intended waste isolation function. Specifically, the WP has been designed to satisfy the appropriate sections of the rule for disposal of HLW at YM (in the current non-site specific Part 60, this is Section 60.135, "Criteria for the waste package and its components").

(4) It is acceptable to use corrosion test results not specifically collected for the YM site provided the results are appropriately interpreted for conditions at the site.

(5) DOE's corrosion testing program is sufficiently complete at the time of the LA submittal to bound likely effects on performance. In addition, DOE's program identifies specific plans for completion of the testing program to reduce any significant areas of uncertainty as part of the performance confirmation program.

4.1.2 Technical Bases for Acceptance Criteria for Subissue 1

Repository regulatory requirements recognize that the engineered barriers provided to isolate radioactive wastes for long periods of time will gradually degrade over that period. It is anticipated that the primary cause for barrier degradation will be some mode, or combination of modes, of material corrosion. Both DOE and the staff have evaluated the most likely forms of barrier materials degradation for the candidate container materials of interest (Farmer, 1988; and U.S. Nuclear Regulatory Commission, 1997). These degradation modes include dry air oxidation of container materials during the initial hot, dry period of repository post-closure performance. Following this initial thermal period, the rock wall and container surface temperatures will decrease, and the container materials will be subjected to humid air corrosion and various modes of aqueous corrosion. These modes include general corrosion, localized corrosion, stress corrosion cracking, galvanic corrosion, microbial corrosion, and hydrogen embrittlement. It is hypothesized that corrosion can initiate, at a time when the relative humidity exceeds a critical value. Under these conditions, it is expected that humid air corrosion will occur in the presence of a thin surface film of condensed fluid which is in contact with the water vapor above the surface. As the temperature continues to decrease, aqueous corrosion will occur as a result of the formation of a thicker film of condensed fluid. The thicker film can also develop from the anticipated influx of water from various thermohydrological processes, such as the heat-pipe effect (Pruess and Tsang 1993), gravity- driven refluxing, and percolation of meteoric water. The relative importance of the various modes of materials degradation and the corresponding impacts on barrier performance are dependent on material selection and the environment interacting with those materials. Resolution of Subissue 1 will necessitate the identification of the most important modes of barrier degradation, as well as numerical estimates of the effects of degradation on container lifetime and radionuclide release from the EBS to the near-field environment.

4.2 Subissue 2: What are the Effects of Materials Stability and Mechanical Failure on the Lifetime of the Containers and the Release of Radionuclides to the Near-Field Environment?

This subissue will be addressed in subsequent revisions of the IRSR.

4.2.1 Acceptance Criteria for Subissue 2

The acceptance criteria will be developed in subsequent revisions of this IRSR.

4.2.2 Technical Bases for Acceptance Criteria for Subissue 2

Technical bases will be described in subsequent revisions of this IRSR.

4.3 Is SF Sufficiently Resistant to Contribute to the Control of Radionuclide Releases to the Near-Field Environment?

This subissue will be addressed in subsequent revisions of this IRSR.

4.3.1 Acceptance Criteria for Subissue 3

The acceptance criteria will be developed in subsequent revisions of this IRSR.

4.3.2 Technical Bases for Acceptance Criteria for Subissue 3

Technical bases will be described in subsequent revisions of this IRSR.

4.4 Is HLW Glass Sufficiently Resistant to Contribute to the Control of Radionuclide Releases to the Near-Field Environment?

This subissue will be addressed in subsequent revisions of this IRSR.

4.4.1 Acceptance Criteria for Subissue 4

The acceptance criteria will be developed in subsequent revisions of this IRSR.

4.4.2 Technical Bases for Acceptance Criteria for Subissue 4

Technical bases will be described in subsequent revisions of this IRSR.

4.5 Review Method for all Subissues

Issue resolution with DOE will be pursued through a continuation of the pre-licensing consultation and interaction that has been ongoing for many years. The staff will review the following: DOE's Site Characterization Progress reports in relation to the further development of container design and materials selection; EBS design documents, such as the "Mined Geologic Disposal System Advanced Conceptual Design Report" (TRW, 1996a) and the planned 1998 Viability Assessment (TRW, 1996b); future repository iterative performance assessments and sensitivity studies; the results of ongoing research and testing on container materials and waste forms; the results of peer reviews or expert elicitations on EBS components; and the results of independent staff analyses, studies, and evaluations of the EBS. Staff will focus its review on the leading candidates for the container materials and their likely modes of degradation, the overall design characteristics or features of the containers, the design basis for the containers, the container fabrication process, as well as the numerical assessments of container performance.

For the waste forms (subissues 3 and 4), the staff will review waste form degradation processes and their comparative releases in both dry air and aqueous environments, including fuel dissolution, HLW glass leaching, formation of secondary minerals and colloids, cladding degradation and mobilization of radionuclides in the EBS.

Numerical assessments will be performed using the most up-to-date versions of the TPA Code. The acceptance criteria will be used to evaluate DOE's demonstration that the containers will be sufficiently long-lived and radionuclide releases will be sufficiently controlled and that EBS will make a significant contribution to overall repository system performance.

5.0 STATUS OF SUBISSUE RESOLUTION AT THE STAFF LEVEL

In prior years, staff raised detailed concerns and questions about the DOE site characterization and PA program in areas related to this KTI. These concerns and questions were documented in the following report:

"NRC Staff Site Characterization Analysis of the U.S. Department of Energy's Site Characterization Plan, Yucca Mountain Site, Nevada," (U.S. Nuclear Regulatory Commission, 1989).

In recent years, the staff recognized the need to refocus the prelicensing repository program on resolving issues most significant to repository performance. The status of the staff's refocused efforts, including general descriptions of the primary issues and concerns in the 10 HLW program subject areas of interest (i.e., 10 KTIs), was documented in the following report:

"NRC High-Level Radioactive Waste Program Annual Progress Report: Fiscal Year 1996" (U.S. Nuclear Regulatory Commission, 1997).

Additional comments and concerns related to the 10 KTIs were documented in the following reports related to DOE's 1995 iterative PA:

NRC/CNWRA Audit Reviews of the DOE TSPA-95 (Austin, 1996a, b, and c); and

NRC/CNWRA "Detailed Review of Selected Aspects of Total System Performance Assessment-1995" (Baca, 1997).

Continuing staff efforts to resolve the issues, concerns, and questions identified in the above reports have resulted in further refinement and clarification of the primary issue and subissues in the CLST subject area of interest, as described in Section 2.0 of this IRSR. In the following sections, a summary is provided on the status of resolution on each of the subissues described in Section 2.0, including the status of the detailed open items resulting from the staff's Site Characterization Analysis (SCA).

5.1 Status of Resolution of Subissue 1 and Related Open Items

Dry Oxidation of Carbon Steel Outer Container

After emplacement, for an extended period that may last several thousand years, the environment in contact with the WPs is expected to be hot and dry air. Under dry conditions corresponding to relative humidities lower than approximately 65 percent, the outer carbon steel container may undergo dry oxidation. Currently, DOE postulates that dry oxidation of the outer container would be negligible for the Mined Geological Disposal System planned at the YM repository site, with predicted metal penetrations of about 2 μm after 10,000 years at 200°C (Stahl, 1993). For this reason, dry oxidation is not considered in the DOE "TSPA-95." More recently, Henshall (1996) has predicted, assuming a parabolic growth law, a general oxidation penetration of 127 μm after exposures to temperatures decreasing from 280 to 210°C over a 5000-year period. Assuming periodic spalling of the oxide, a general penetration of 350 to 600 μm was estimated over 5000 years.

A review of dry oxidation, focused predominantly on iron-base alloys with varying alloy content (Ahn, 1996c), revealed that at temperatures above 600°C, iron-base alloys often show localized dry oxidation (Shida and Moroishi, 1992; Otsuka and Fujikawa, 1991; Newcomb and Stobbs, 1991; Tasovac, et al., 1989; Mayer and Smeltzer, 1973; Raman, et al., 1992). This localized oxidation is normally much deeper than general oxidation. The extrapolated values suggest shallow (at most 100 µm at 200°C for 10,000 years) penetration by localized dry oxidation. However, if oxygen transport along grain boundaries is assumed to be the rate controlling step for intergranular oxidation of these alloys, penetrations deeper than calculated above can result because grain boundary diffusivity is generally higher than lattice diffusivity for substitutional elements, such as oxygen. If one uses a classical Arrhenius relation and assumes a preexponential factor several orders of magnitude larger and an activation energy close to one half for grain boundary diffusion compared to the same parameters for lattice diffusivity, oxygen penetration can occur through the 10-cm container wall in less than 10,000 years at both 150 and 200°C (Ahn, 1996c). The *consequences of this penetration may include: (i) container breach, and (ii) easy mechanical failure* of container by localized oxidation or by (atomic or gaseous) oxygen embrittlement.

However, for the C-Mn steels proposed in DOE design of the outer overpack, oxygen transport has not been shown to be the rate controlling process at the temperatures of interest. Larose and Rapp (1997) undertook a detailed examination of the potential for dry oxidation at repository temperatures, with a specific focus on C-Mn and low-alloy steels. As a baseline, the thermodynamics and kinetics of pure iron oxidation were considered. Thermodynamics of the Fe-O system, from room temperature to 1600°C, was used to understand the oxide phases important within the small temperature range of interest to the repository. The literature on oxidation of iron or steel at temperatures beyond 567°C is not considered relevant because the stable oxide phase at low oxygen fugacities beyond this temperature is wüstite, corresponding to Fe_xO, which is slightly deficient in Fe (Muan and Osborne, 1965). At lower temperatures, the oxide scale on iron has two phases--an inner magnetite (Fe_3O_4) and an outer hematite (Fe_2O_3) phase. The kinetics of oxide growth at temperatures below 567 °C are dictated by cation diffusion outwards not by oxygen transport inwards. Grain boundaries in the oxide scale are known to influence the iron diffusivities in oxides, but the values are not known at the temperatures of interest. Oxidation of pure iron between 400 and 550°C is especially affected by delamination of the scale due to condensation of cation vacancies at the metal-oxide interface. At 250°C, carbon steels containing 0.2 weight percent carbon are expected to lose about 4 µm in thickness, while steels containing 2.25 weight percent chromium and 1 weight percent molybdenum (another candidate container material) are expected to undergo a thickness loss of about 3 µm in 1000 years at the same temperature. The oxidation rate of steel at low temperatures increases with the carbon content in steel. They concluded that because low-temperature oxidation in carbon and low-alloy steels following a parabolic rate law is controlled by outward diffusion of iron rather than inward diffusion of oxygen, intergranular penetration of oxide would not be significant.

Based on the arguments above, it can be concluded, at least for C-Mn steels, that dry oxidation is not a significant failure process. Accordingly, the staff considers the issue related to the potential significance of dry oxidation as a failure process to be resolved. In this regard, issue resolution at the staff level, during prelicensing, is achieved when the staff has no more comments or questions at a point in time in relation to DOE's program for addressing an issue. However, if long-term exposure of carbon steel samples coated with various ceramic materials and humid-air corrosion tests (DOE, 1997) yield evidence of incipient intergranular oxidation, through cross sectional examination by high-magnification metallography, the issue of dry oxidation will be reexamined both in terms of modeling and experimental work.

Dry Oxidation Component of Subissue 1

Two open items identified in NRC staff's SCA are related to dry oxidation: Comment 85 and Question 49 (U.S. Nuclear Regulatory Commission, 1989). The staff noted in the comment that corrosion (in this case, dry oxidation) could contribute to wall thinning for the design lifetime of the container.

Type: Comment 85
Source: SCA
Status: Closed
Title: No consideration of temporal changes in the state of stress on WP corrosion

DOE states that uniform penetration by dry oxidation--regardless of oxide spallation, which accelerates the uniform penetration--is insignificant (Stahl, 1993; and Henshall, 1996). NRC reviewed the mechanism of dry oxidation and has determined that the container is unlikely to fail by that mechanism (Ahn, 1996c; and Larose and Rapp, 1997), despite a small possibility of intergranular diffusion of oxygen and concurrent intergranular oxidation. Therefore, NRC takes the position that the matter of dry oxidation associated with the open item is closed. NRC notes that current and long-term Lawrence Livermore National Laboratories (LLNL) tests may yield new information on whether intergranular oxidation will take place. If intergranular diffusion of oxygen is able to be ruled out by the LLNL experiments, the amount of wall thinning due to dry oxidation can be more definitely concluded to be insignificant for the container life.

Item ID: Question 49
Source: SCA
Status: Closed
Title: Effects of low temperature oxidation on containers

Resolution of Question 49, in regard to dry oxidation, is considered achieved on the basis of information from DOE indicating that DOE's corrosion test plan includes evaluation of the effects of surface conditioning by dry oxidation. It is anticipated that such an evaluation will include a determination of whether or not dry oxidation has significant effects on other corrosion modes under Subissue 1 (localized corrosion, stress corrosion cracking, and galvanic corrosion).

5.2 Status of Resolution of Subissue 2 and Related Open Items

5.3 Status of Resolution of Subissue 3 and Related Open Items

5.4 Status of Resolution of Subissue 4 and Related Open Items

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APPENI Y

Status of U.S. Nuclear Regulatory Commission Site Characterization Analysis Open Items on Waste Package and Release from Engineered Barrier System

Item ID	Source	Title	Status	KTI	Comment
Comment 5	SCA	interpretation of substantially complete containment	resolved 7/11/94	CLST	
Comment 25	SCA	rationale on additional testing on waste and interactions between and among radionuclides on sorption		CLST ENFE RT	
Comment 44	SCA	overall goal is not consistent with substantially complete containment	resolved 7/31/91	CLST	
Comment 4	SCA	relationship of postclosure tectonics to the waste package and the engineering barrier system requirement		SDS CLST	
Comment 79	SCA	adequacy of waste package corrosion tests for the repository		ENFE CLST	
Comment 80	SCA	performance goals consistent with interpretation and intent of substantially complete containment	resolved 3/7/95	CLST	
Comment 81	SCA	adequacy of program in stress corrosion cracking behavior of waste packages		CLST	
Comment 82	SCA	there is an inadequate discussion on how the waste package performance may be verified at the time of license application		CLST	
Comment 83	SCA	the term "uniform corrosion" is misleading	resolved 7/31/91	CLST	
Comment 84	SCA	issue resolution strategy and testing package for the waste package and engineering barrier system do not take into account the full range of likely natural conditions that might affect performance of the barrier		CLST SDS	
Comment 85	SCA	performance assessment: temporal changes in the state of stress due to corrosion of the container is not accounted for	resolved 3/9/98	CLST	

Comment 86	SCA	degradation modes of copper-based alloys do not appear to agree with scientific literature		CLST	
Comment 87	SCA	adequacy of effects of dissimilar metal contacts causing corrosion		CLST	
Comment 88	SCA	assumption of reduced uncertainties because of the unsaturated zone		CLST	
Comment 89	SCA	construction materials may change the local pH and affect the corrosion of the metal containers and the leach rates of radionuclides from the glass		ENFE CLST	
Comment 90	SCA.	consideration of varying oxygen concentrations on the corrosion of metal containers		CLST ENFE	
Comment 91	SCA	waste package/Performance Assessment: Consideration of alternative canisters for carbon-14 releases		TSPAI CLST	
Comment 118	SCA	the monitoring and testing activities should include long-term in situ and long-term waste package activities		TSPAI CLST	
Question 30	SCA	water quality as related to waste package design		ENFE USFIC CLST	
Question 31	SCA	integrity of spent fuel cladding		CLST	
Question 32	SCA	container "similarity" for borosilicate glass waste vs. spent fuel		CLST	
Question 33	SCA	justification for accepting 5L [illegible]f accumulated standing wate[illegible] [illegible]er canister in the first 1000 years		RDTME CLST ENFE	
Question 34	SCA	meaning of "undetected defective closures" in waste package fabrication and handling design goals		CLST	
Question 35	SCA	acceptance criteria for helium leak results	resolved 3/7/95	CLST	
Question 36	SCA	explanation and justification for use of corrosive surface finishing chemicals on waste package prior to emplacement		CLST	

Question 37	SCA	basis for 10 cm or more of free fall for canister and contents	resolved 3/7/95	CLST	
Question 38	SCA	basis for 1mm scratch criterion to avoid emplacement of damaged canisters		CLST	
Question 39	SCA	meaning of "unusual process history" as a criterion to avoid emplacement of damaged canisters		CLST	
Question 40	SCA	basis for using a factor of 2 for corrosion for rate for borehole liner in comparison to container material		CLST	
Question 43	SCA	anticipated operational occurrences considered part of normal conditions on the preclosure design and analysis	resolved 7/31/91	SDS CLST	
Question 44	SCA	basis for assumed numbers of breached assemblies or canisters		TSPAI CLST	
Question 46	SCA	basis for stricter containment of long half-life isotopes	resolved 7/11/94	CLST	
Question 47	SCA	what is the origin of the stated definition of a container failure	resolved 3/7/95	CLST	
Question 48	SCA	selection of peer review panel on waste package	resolved 7/31/91	CLST	
Question 49	SCA	effects of low temperature oxidation on containers	resolved 3/9/98	CLST	
Question 50	SCA	assumption that stress propagation results in corrosion	resolved 7/31/91	CLST	
Question 51	SCA	impacts of INEL and Hanford high-level wastes on the YM Program	resolved 11/8/94	CLST	
Question 52	SCA	leaching properties specification will require the producer to control leaching characteristics of the glass waste	resolved 7/31/91	CLST	
Question 53	SCA	specification of cooling rate of the glass waste	resolved 3/7/95	CLST	
Question 54	SCA	release rates of radionuclides from spent fuels in J-13 water	resolved 7/31/91	CLST	