Department of Energy
Washington, DC 20585

July 1, 1994

Mr. Joseph J. Holonich, Chief
High-Level Waste and Uranium
Recovery Project Branch
Division of Waste Management
Office of Nuclear Material
Safety and Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

References: (1) Ltr, Shelor to Linehan, dtd 12/14/90
(2) Ltr, Bernero to Bartlett, dtd 7/31/91
(3) Ltr, Shelor to Holonich, dtd 6/10/94
(4) Ltr, Roberts to Holonich, dtd 2/3/93

Dear Mr. Holonich:

On December 14, 1990, the U.S. Department of Energy (DOE) transmitted its responses to objections, comments, and questions presented in the U.S. Nuclear Regulatory Commission's (NRC) Site Characterization Analysis (SCA) (Reference 1). The NRC staff evaluated these responses, closing some of the items and creating open items of the remainder (Reference 2). Three of the open items have been addressed through actions and progress in the program. Enclosures 1-3 state the administrative records with respect to SCA Questions 35, 45, and 51. DOE believes that responses to these open items provided in this letter are sufficient to close them, and awaits NRC confirmation.

Questions 35, 45, and 51 pertain to waste package design and waste package failure modes, and are briefly summarized below:

Comment 35 asks if the acceptance criteria for a waste package helium leak test is consistent with the performance requirements of 10 Code of Federal Regulations Part 60.113 for the engineered barrier system.

DOE's basis for resolution indicates that the new performance goal to achieve mean waste package lifetimes well in excess of 1000 years will provide confidence in containment of radionuclides as explained in DOE's approach to resolve the "substantially complete containment" issue (Reference 3).

Comment 45 asks what site characterization plans are in place to study waste package failure modes in the area of particulate sources terms, retention factors, plateout, and gravitational settlement factors.

DOE's basis for resolution is to report the process used to identify if any plans are needed. A preliminary preclosure safety analysis has been completed by DOE where conservative estimates for particle distributions involving accidents during transport in the operations area and particle egress from breached containers have been made. The estimates are derived from assessments for

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accidents in reactor cores, transfer cells, and dry storage sites. DOE has no specific plans to study particle size distributions for spent nuclear fuel/high-level defense waste repository accidents at this time. The need for design studies to obtain data or perform analyses for such an assessment would result from the analyses conducted for Determination of Importance Evaluations involving the systems and components for waste package transfer, transport, and emplacement operations.

Comment 51 asks if DOE has considered impacts to waste package design with respect to the Idaho National Engineering Laboratory and Hanford defense waste forms.

DOE's basis for resolution is to identify that the process for accepting nonstandard or alternative waste forms that are not now under the purview of the Civilian Radioactive Waste Management program. These waste forms, such as those that may arise from different parts of the DOE complex, are specified in the DOE Waste Acceptance System Requirements Document. This document has been previously sent to the NRC (Reference 4).

The DOE believes that resolution of these open items is important to proceed with repository advanced conceptual design. DOE requests that a review and response to this letter be made available to DOE by September 15, 1994.

If you have any questions, please contact Chris Einberg of my staff at (202) 586-8869.

Sincerely,

Dwight E. Shelor
Associate Director for
Systems and Compliance
Office of Civilian Radioactive
Waste Management

Enclosures:
1. Administrative Record for SCA Question 35
2. Administrative Record for SCA Question 45
3. Administrative Record for SCA Question 51

cc: w/enclosures:

R. Nelson, YMPO
R. Loux, State of Nevada
W. Offutt, Nye County, NV
T. J. Hickey, Nevada Legislative Committee
D. Bechtel, Las Vegas, NV
Eureka County, NV
Lander County, Battle Mountain, NV
P. Niedzielski-Eichner, Nye County, NV
L. Bradshaw, Nye County, NV
C. Schank, Churchill County, NV
F. Mariani, White Pine County, NV
V. Poe, Mineral County, NV
J. Pitts, Lincoln County, NV
J. Hayes, Esmeralda County, NV
B. Mettam, Inyo County, CA

Enclosure 1

SCA Question 35 and Original DOE Response

NRC Evaluation of Original DOE Response

DOE Supplemental Response to NRC Question 35

ENCLOSURE 1

Section 8.3.4.2.G. Waste package fabrication and handling before emplacement.
Design goal for closure, p. 8.3.4.2-30 para. 6.

QUESTION 35

It is stated that the closure process will be capable of being performed and inspected under remote conditions with a reliability such that the containment would be capable of passing a standard helium leak test at the level of 1 x 10^{-7} atm-cm^3/sec.

What is the basis for the helium leak tes' acceptance criteria?

BASIS

10 CFR Part 60.113 includes requirements for the performance of the engineered barrier system and it is not clear if the criteria are consistent with these requirements.

RECOMMENDATION

Provide the basis for the helium leak test acceptance criteria and demonstrate that the criteria are consistent with the performance requirements of 10 CFR Part 60.113 for the engineered barrier system.

RESPONSE

Design goal for closure. The closure process will be capable of reliable remote operation to seal the containers as required. A preliminary definiticn of sealed is passing a standard helium leak test (such as ASME Section V, Article 10, Appendix IV, 1986) to a level of $1x10^7$ atm-cm^3/s. This goal will be assessed further during waste package design.

The closure inspection process will be able to assure that the container is sealed and will have a high reliability for detecting design limit flaws. The preliminary reliability is set at 99% or greater. The design limit flaws have not been determined. These goals would be assessed further during waste package design.

Section 8.3.4.2.G Waste package fabrication and handling before emplacement. Design goal for closure, p. 8.3.4.2-30 para. 6

SCA QUESTION 35

It is stated that the closure process will be capable of being performed and inspected under remote conditions with a reliability such that the containment would be capable of passing a standard helium leak test at the level of 1 x 10E-7 atm-cu cm/sec.

What is the basis for the helium l ak test acceptance criteria?

EVALUATION OF DOE RESPONSE

- DOE cites ASME Section V, Article 10, Appendix IV, 1986 as the basis for the helium leak test acceptance criteria and indicates that the criteria will be assessed further during waste package design. However, DOE does not provide any assessment or information that demonstrates that the performance requirements of 10 CFR 60.113 for the engineered barrier system.

- The NRC staff considers this question closed as to the basis for the helium leak test acceptance criteria, but open as to whether the criteria are consistent with 10 CFR 60.113.

DOE Supplemental Response to Question 35

The definition of "substantially complete containment" was addressed in DOE's package of supplemental responses to Comments 5 and 80. In those responses, DOE stated that a new performance goal has been established which focuses on containment of radionuclides. The goal is to achieve mean waste package lifetimes well in excess of 1000 years. This means that the number of failures at the initial tail of the failure distribution over time, i.e., during the containment period, will be very small. The DOE will achieve this performance goal through the use of multiple barriers with more than one failure mode. This permits the peak of the failure distribution of the combined waste package to be reduced and the distribution itself to be extended in time. Thus, the fraction failed at 1,000 years will be extremely small, on the order of 1%. This new approach, which focuses on containment, is consistent with the NRC's emphasis on containment rather than release during the containment period.

Enclosure 2

SCA Question 45 and Original DOE Response

NRC Evaluation of Original DOE Response

DOE Supplemental Response to NRC Question 45

ENCLOSURE 2

Section 8.3.5.5.1 Information Need 2.3.1: Determination of credible accident sequences and their respective frequencies applicable to the repository.

QUESTION 45

The SCP does not identify whether additional data are needed to establish particulate source terms for the waste package, particulate retention factors by containing vessels, or plateout or gravitational settlement factors for the geologic repository operations area during accident conditions in the preclosure phase. What investigations are planned?

BASIS

o This question, which was originally posed as CDSCP Question 44, is repeated here since no changes or additions were made to the SCP in response to the question.

o Several statements in Sections 5.1.2-5.1.5 of the CDR seem to indicate that better bases for waste package source terms and releases from the geologic repository operations area are needed.

o The SCP does not discuss the need for investigations to characterize the magnitude (or particle sizes) of radionuclides that could be released from the waste package when subjected to impacts (such as a crane falling on a fuel assembly) nor does it discuss the need for investigations to develop realistic radionuclide retention fractions for containment systems and structures.

RECOMMENDATIONS

o Existing information on the source terms for the waste package and plateout and retention factors for the geologic repository operations area in the preclosure phase needs to be evaluated in SCP updates and the need (if any) for additional information (e.g. data gathering, models, etc.) to be obtained during site characterization needs to be identified.

o If new information is to be obtained, the investigations should be discussed in SCP updates.

REFERENCES

H.R. MacDougall, L.W. Scully, and J.R. Tillerson, "Nevada Nuclear Waste Storage Investigations Project, Site Characterization Plan Conceptual Design Report," SAND84-2641, Volume 4, Appendix F, September 1987: Section 5.1.5 Release Factors for Gap Radioactivity; Section 5.1.3 Fuel Pellet and HLW Glass Pulverization Factors; Section 5.1.4 Particulate Retention Factors for Fuel Cladding, Casks, DHLW Canister, and Waste Disposal Containers; Section 5.1.5 Particulate Retention Factors by Building and Hot Cells.

P.A. Harris, D.M. Ligon, and M.G. Stamatelatos, GA Technologies, Inc., "High-Level Waste Preclosure Systems Safety Analysis, Phase I, Final Report," USNRC Report NUREG/CR-4303 (July 1985).

RESPONSE

The U.S. Department of Energy recognizes that additional data are needed to establish particulate source terms for the waste package, particulate retention factors by containing vessels, or gravitational settlement factors for the geologic repository operations area. However, the Site Characterization Plan is intended specifically to identify site data needed and as such these data are not included as part of site characterization. Plans for investigations to collect these data would be developed as part of the supporting studies and analyses planned during the next phase of repository design.

Section 8.3.5.5.1 Information Need 2.3.1: Determination of credible accident sequences and their respective frequencies applicable to the repository

SCA QUESTION 45

The SCP does not identify whether additional data are needed to establish particulate source terms for the waste package, particulate retention factors by containing vessels, or plateout or gravitational settlement factors for the preclosure phase. What investigations are planned?

EVALUATION OF DOE RESPONSE

- In its response, DOE recognized a future need for additional design data of this type, but its response did not describe any plans for the investigations necessary to obtain this data.

- DOE's response states that the data in question are design data rather than site data and that as such they are not included as part of site characterization. The NRC staff disagrees with this interpretation of site characterization.

- The NRC staff considers this question open. To close this question, DOE needs to identify its plans for investigations of these phenomena.

DOE Supplemental Response to Question 45

DOE believes that spent nuclear fuel/high-level defense waste (SNF/HLDW) particle generation and the attendant size distribution are not (MGDS) site-specific problems (i.e., site characterization), but are design questions associated with nuclear waste activities. The fraction of particles generated by an accident involving a spent fuel assembly at a reactor site, in a transfer cell, or at a dry storage site has been estimated for the assessment of the consequences of design basis accidents for these currently operating systems.

A preliminary MGDS/ESF preclosure safety analysis has recently been completed. This investigation, based on the available literature, considered two initiating events (rock falls and waste transporter accidents) that could result in accidental releases of radionuclides to the accessible environment. Assessing the dose consequences for these accidents required that estimates be made for SNF/HLDW particulate generation from accidental energetic encounters, particle retention in containment systems and structures, and the mitigating effects of plateout and fallout (gravitational settling). Adequate detail was found on the plateout and fallout processes to allow reasonable yet conservative estimates to be made for both particle attrition during transport in the operations areas and particle egress from breached waste containers. However, as previously identified in Question 45 by both the DOE and NRC, the fraction of solid SNF/HLDW converted to particulate matter of respirable size as a result of an energetic event appears to be not well defined. The issue of particle generation for an MGDS is to be addressed by analyses for Determination of Importance Evaluations for the systems and components for waste package transfer, transport, and emplacement operations. Potential data needs are being reviewed by performance assessors. This review will incorporate some sensitivity and uncertainty analysis of the dose consequence for each process, to ensure effort is deployed in those areas that have the larger impact on dose consequences.

Enclosure 3

SCA Question 51 and Original DOE Response

NRC Evaluation of Original DOE Response

DOE Supplemental Response to NRC Question 51

ENCLOSURE 3

Section 8.3.5.10 Issue resolution strategy for Issue 1.5: Will the waste package and repository engineered barrier system meet the performance objective for radionuclide release rates as required by 10 CFR 60.113?

Section 7.3.1.1.2 High-level wastes

Section 7.4.3.2 Glass waste form performance research

QUESTION 51

Has DOE con idered the impacts to the waste package site characterization program related to INEL and Hanford high-level wastes?

BASIS

o Section 7.3.1.1.2 discusses receipt of high-level wastes from the West Valley Demonstration Project (WVDP) and from the Defense Waste Processing Facility (DWPF). High-level wastes from INEL and Hanford are not mentioned.

o Section 7.4.1.1.2 discusses waste form research addressing wastes from WVDP and DWPF but does not mention research addressing INEL and Hanford wastes.

o High-level liquid waste generated at INEL by the processing of spent fuel from the national defense (naval propulsion nuclear reactors) and reactor testing programs and by the reprocessing of fuel from nondefense research reactors is stored in large, doubly contained, underground stainless steel tanks. The liquid waste is converted to a calcine, then stored underground in stainless steel bins housed in reinforced concrete vaults. The INEL wastes are acidic.

o The Hanford waste was generated by reprocessing of production reactor fuel for recovery of plutonium, uranium, and neptunium for defense and other federal programs. Most of the high-heat-emitting isotopes (90Sr and 137Cs) have been removed from the waste, converted to solid strontium fluoride and cesium chloride, placed in double-walled capsules, and stored in water basins. The liquid sludge, slurry, and salt calk are stored in underground concrete tanks with carbon-steel liners. The Hanford wastes are alkaline.

o The total volume of unprocessed wastes from INEL and Hanford is approximately 500 thousand cubic meters which is much larger than the 115 thousand cubic meters of DWPF and WVDP wastes (DOE/RE-0017/2).

RECOMMENDATIONS

o Include discussions of INEL and Hanford wastes in the SCP.

o Examine the quantity and characteristics of wastes from INEL and Hanford and plans for ultimate disposition, consider their impact on SCP planning and tests, and make appropriate changes to plans and tests.

REFERENCES

DOE/NE-0017/2, Spent Fuel and Radioactive Waste Inventories, Projections, and Characteristics, September, 1983

RESPONSE

The report "Evaluation and Selection of Borosilicate Glass as the Waste Form for Hanford High Level Radioactive Waste" addresses the selection of borosilicate glass for the Hanford high-level waste. Since Savannah River and West Valley waste forms are borosilicate glass, the Hanford high-level waste in a borosilicate glass should have no additional impacts to the waste package site characterization program.

A selection for the waste form at Idaho National Engineering Laboratory (INEL) has not been made. After additional information and selection of the waste form (glass-ceramics is one being studied) has been provided, the impact on the waste package site characterization program would be assessed.

REFERENCES:

DOE (U. S. Department of Energy), 1990. Evaluation and Selection of Borosilicate Glass as the Waste Form for Hanford High Level Radioactive Waste, DOE/RL-90-27, Richland Operations Office, Richland, WA.

Section 8.3.5.10	Issue resolution strategy for Issue 1.5: Will the waste package and repository engineered barrier system meet the performance objective for radionuclide release rates as required by 10 CFR 60.113?
Section 7.3.1.1.2	High-level wastes
Section 7.4.3.2	Glass waste form performance research

SCA QUESTION 51

Has DOE considered the impacts to the was' : package site characterization program related to Idaho National Engineering Laboratory (INEL) and Hanford high-level wastes?

EVALUATION OF DOE RESPONSE

- o DOE cites report DOE/RL-90-27 (1990) as the basis for the selection of borosilicate glass for the Hanford high-level wastes. However, DOE does not discuss how the quantity and characteristics of Hanford wastes might impact SCP planning and tests and ultimate disposition.
- o DOE indicates that it will assess the impact of Idaho National Engineering Laboratory high-level wastes after additional information and selection of the waste form for those wastes has been made.
- o The NRC staff considers this question open.

REFERENCES

DOE (U.S. Department of Energy), 1990. Evaluation and Selection of Borosilicate Glass as the Waste Form for Hanford High Level Radioactive Waste, DOE/RL-90-27, Richland Operations Office, Richland, WA.

DOE Supplemental Response to Question 51

The DOE has considered the impacts of Idaho National Engineering Laboratory (INEL) and Hanford high-level waste. The potential number of waste canisters from these sources, based on available information, has been factored into the waste stream analysis for the potential repository at Yucca Mountain. This information is shown in the Waste Acceptance-System Requirements Document (WA-SRD) (DOE/RW-0315, Revision 1). The number of high-level waste canisters containing glass has been factored into the design of the waste packages and the layout of the repository.

The WA-SRD addresses the acceptance of standard waste forms, spent nuclear fuel, and high-level waste glass into the waste management system. If the INEL and Hanford high-level waste forms are the standard borosilicate glass waste form, no additional effort will be required. However, the producers are evaluating alternative, non-standard waste forms. The WA-SRD also addresses the issue on non-standard waste forms, should they be developed by INEL or Hanford. The document describes how the acceptance criteria will be modified to include these waste forms if they are proposed for acceptance into the civilian waste management system. Any change proposal will be generated by DOE/Environmental Restorative and Mitigation Program and will be accompanied by the scientific basis. The scientific basis will be confirmed by independent research by the Office of Civilian Radioactive Waste Management (OCRWM). Thus, OCRWM has in place plans to evaluate the performance of these alternate waste forms once the character of these waste forms is better defined. The evaluation of these alternate waste forms will follow the studies detailed in the Waste Package Implementation Plan (YMP/92-11 Revision 0, ICN 2).