March 20, 2003

Joseph D. Ziegler, Acting Director
Office of License Application and Strategy
U.S. Department of Energy
Office of Repository Development
P.O. Box 364629 M/S 523
North Las Vegas, NV 89036-8629

SUBJECT: CONTAINER LIFE AND SOURCE TERM KEY TECHNICAL ISSUE AGREEMENT CLST.2.03; STATUS: NEEDS ADDITIONAL INFORMATION

Dear Mr. Ziegler:

During a Technical Exchange and Management Meeting held on September 12-13, 2000, the U.S. Nuclear Regulatory Commission (NRC) and the U.S. Department of Energy (DOE) reached agreement on a number of issues within the Container Life and Source Term (CLST) Key Technical Issue (KTI). Subsequently, during a Technical Exchange and Management Meeting held on April 15-16, 2002, DOE indicated that the information requested in CLST 2.03, rather than being provided in Analysis and Model Reports, would be submitted early in the form of a Letter Report. By letter dated September 27, 2002, DOE submitted information in an attached Letter Report to address CLST 2.03. After an initial review by NRC it was determined that the referenced report "Comparison of the Traditional Strength of Materials Approach to Design with the Fracture Mechanics Approach," CAL-EBS-ME-000019 Rev. 00, would need to be reviewed in order to adequately evaluate the DOE Letter Report. The NRC staff has reviewed this information, with respect to the agreement, and the results of the staff's review are enclosed.

The NRC has reviewed the DOE report addressing KTI agreement item CLST 2.03. In general, the approach used by DOE to establish the governing failure mechanism (i.e., brittle fracture or plastic collapse) for the drip shield and waste package materials using failure assessment diagrams is appropriate. The response to the agreement, however, failed to adequately establish the appropriate failure mechanisms for the drip shield and waste package materials because the various material properties used in the analyses have yet to be satisfactorily justified. To accomplish this task, additional information is needed by the NRC staff. Therefore, NRC staff, as indicated in the attached, lists CLST Agreement 2.03 as needing more information.

If there are any questions regarding this letter, please contact Daniel Rom at (301) 415-6704 or by e-mail at dsr@nrc.gov.

Sincerely,

/RA/

Janet R. Schlueter, Chief
High-Level Waste Branch
Division of Waste Management
Office of Nuclear Material Safety
and Safeguards

Enclosure: As stated
cc: See attached distribution list

Letter to J. Ziegler from J. Schlueter dated March 20, 2003
cc:

A. Kalt, Churchill County, NV	M. Corradini, NWTRB
R. Massey, Churchill/Lander County, NV	J. Treichel, Nuclear Waste Task Force
I. Navis, Clark County, NV	K. Tilges, Shundahai Network
E. von Tiesenhausen, Clark County, NV	M. Chu, DOE/Washington, D.C.
G. McCorkell, Esmeralda County, NV	G. Runkle, DOE/Washington, D.C.
L. Fiorenzi, Eureka County, NV	C. Einberg, DOE/Washington, D.C.
A. Johnson, Eureka County, NV	S. Gomberg, DOE/Washington, D.C.
A. Remus, Inyo County, CA	W. J. Arthur, III , DOE/ORD
M. Yarbro, Lander County, NV	R. Dyer, DOE/ORD
L. Stark, Lincoln County, NV	C. Newbury, DOE/ORD
M. Baughman, Lincoln County, NV	J. Ziegler, DOE/ORD
L. Mathias, Mineral County, NV	A. Gil, DOE/ORD
L. Bradshaw, Nye County, NV	W. Boyle, DOE/ORD
D. Chavez, Nye County, NV	D. Williams, DOE/ORD
D. Hammermeister, Nye County, NV	D. Brown, DOE/OCRWM
J. Larson, White Pine County, NV	S. Mellington, DOE/ORD
J. Ray, NV Congressional Delegation	C. Hanlon, DOE/ORD
B. J. Gerber, NV Congressional Delegation	T. Gunter, DOE/ORD
F. Roberson, NV Congressional Delegation	S. Morris, DOE/ORD
T. Story, NV Congressional Delegation	K. Mitchell, BSC
J. Reynoldson, NV Congressional Delegation	D. Krisha, BSC
L. Hunsaker, NV Congressional Delegation	S. Cereghino, BSC
S. Joya, NV Congressional Delegation	N. Williams, BSC
K. Kirkeby, NV Congressional Delegation	M. Voegele, BSC/SAIC
R. Loux, State of NV	D. Beckman, BSC/B&A
S. Frishman, State of NV	W. Briggs, Ross, Dixon & Bell
S. Lynch, State of NV	P. Johnson, Citizen Alert
M. Paslov Thomas, Legislative Counsel Bureau	R. Holden, NCAI
J. Pegues, City of Las Vegas, NV	B. Helmer, Timbisha Shoshone Tribe
M. Murphy, Nye County, NV	R. Arnold, Pahrump Paiute Tribe

cc: (Continued)

R. Clark, EPA

F. Marcinowski, EPA

R. Anderson, NEI

R. McCullum, NEI

S. Kraft, NEI

J. Kessler, EPRI

D. Duncan, USGS

R. Craig, USGS

W. Booth, Engineering Svcs, LTD

E. Opelski, NQS

L. Lehman, T-REG, Inc.

S. Echols, ESG

A. Bacock, Big Pine Paiute Tribe of the Owens Valley

H. Blackeye, Jr., Duckwater Shoshone Tribe

M. Smurr, BNFL, Inc.

T. Kingham, GAO

D. Feehan, GAO

E. Hiruo, Platts Nuclear Publications

C. Anderson, Las Vegas Paiute Tribe

R. Boland, Timbisha Shoshone Tribe

J. Birchim, Yomba Shoshone Tribe

C. Meyers, Moapa Paiute Indian Tribe

V. Miller, Fort Independence Indian Tribe

M. Bengochia, Bishop Paiute Indian Tribe

J. Egan, Egan & Associates, PLLC

J. Leeds, Las Vegas Indian Center

R. Bahe, Benton Paiute Indian Tribe

C. Bradley, Kaibab Band of Southern Paiutes

R. Joseph, Lone Pine Paiute-Shoshone Tribe

L. Tom, Paiute Indian Tribes of Utah

E. Smith, Chemehuevi Indian Tribe

J. Charles, Ely Shoshone Tribe

D. Crawford, Inter-Tribal Council of NV

R. Quintero, Inter-Tribal Council of NV (Chairman, Walker River Paiute Tribe)

D. Eddy, Jr., Colorado River Indian Tribes

H. Jackson, Public Citizen

J. Wells, Western Shoshone National Council

R. Henning, BSC

I. Zabarte, Western Shoshone National Council

March 20, 2003

Joseph D. Ziegler, Acting Director
Office of License Application and Strategy
U.S. Department of Energy
Office of Repository Development
P.O. Box 364629 M/S 523
North Las Vegas, NV 89036-8629

SUBJECT: CONTAINER LIFE AND SOURCE TERM KEY TECHNICAL ISSUE AGREEMENT CLST.2.03; STATUS: NEEDS ADDITIONAL INFORMATION

Dear Mr. Ziegler:

During a Technical Exchange and Management Meeting held on September 12-13, 2000, the U.S. Nuclear Regulatory Commission (NRC) and the U.S. Department of Energy (DOE) reached agreement on a number of issues within the Container Life and Source Term (CLST) Key Technical Issue (KTI). Subsequently, during a Technical Exchange and Management Meeting held on April 15-16, 2002, DOE indicated that the information requested in CLST 2.03, rather than being provided in Analysis and Model Reports, would be submitted early in the form of a Letter Report. By letter dated September 27, 2002, DOE submitted information in an attached Letter Report to address CLST 2.03. After an initial review by NRC it was determined that the referenced report “Comparison of the Traditional Strength of Materials Approach to Design with the Fracture Mechanics Approach,” CAL-EBS-ME-000019 Rev. 00, would need to be reviewed in order to adequately evaluate the DOE Letter Report. The NRC staff has reviewed this information, with respect to the agreement, and the results of the staff’s review are enclosed.

The NRC has reviewed the DOE report addressing KTI agreement item CLST 2.03. In general, the approach used by DOE to establish the governing failure mechanism (i.e., brittle fracture or plastic collapse) for the drip shield and waste package materials using failure assessment diagrams is appropriate. The response to the agreement, however, failed to adequately establish the appropriate failure mechanisms for the drip shield and waste package materials because the various material properties used in the analyses have yet to be satisfactorily justified. To accomplish this task, additional information is needed by the NRC staff. Therefore, NRC staff, as indicated in the attached, lists CLST Agreement 2.03 as needing more information.

If there are any questions regarding this letter, please contact Daniel Rom at (301) 415-6704 or by e-mail at dsr@nrc.gov.

Sincerely,
/RA/
Janet R. Schlueter, Chief
High-Level Waste Branch
Division of Waste Management
Office of Nuclear Material Safety and Safeguards

Enclosure: As stated
cc: See attached distribution list

DISTRIBUTION:

File Center	DWM r/f	HLWB r/f	EPAB r/f	LCampbell	Dhiggs	TAhn
JBradbury	NColeman	PJustus	CMcKenney	TMcCartin	MNataraja	JPohle
JTrapp	SWastler	DBrooks	ACNW	Center	LSN	PUBLIC
JFirth	TBloomer					

S:\DWM\HLWB\TEB\CLST Agreement Letter - 203.wpd.

OFC	HLWB	HLWB	HLWB	DWM	HLWB
NAME	DRom	TBloomer	KStablein	TMcCartin	JSchlueter
DATE	03/11/03	03/11/03	03/11/03	03/20/03	03/20/03

OFFICIAL RECORD COPY

NRC Review of DOE Documents Pertaining to Key Technical Issue Agreements

The U.S. Nuclear Regulatory Commission (NRC) goal of issue resolution during the prelicensing period is to ensure that the U.S. Department of Energy (DOE) has assembled enough information on a given issue for NRC to accept a license application for review. Resolution by the NRC staff during prelicensing does not preclude anyone from raising any issue for NRC consideration during the licensing proceedings. Furthermore, resolution by the NRC staff during prelicensing does not prejudge what the NRC staff evaluation of that issue will be after its licensing review. Issues are resolved by the NRC staff during pre-licensing when the staff has no further questions or comments about how DOE is addressing an issue. Pertinent new information could raise new questions or comments on a previously resolved issue.

This enclosure addresses NRC and DOE Agreement CLST 2.03, which was made during the Container Life and Source Term (CLST) Technical Exchange and Management Meeting (see NRC letter dated October 4, 2000, which summarized the meeting). By a letter dated September 27, 2002, DOE submitted information to address CLST Agreement 2.03. The information submitted for this agreement is discussed below.

Container Life and Source Term Agreement CLST.2.03

Wording of the Agreement: Demonstrate how the Tresca failure criterion bounds a fracture mechanics approach to calculating the mechanical failure of the drip shield. DOE stated that it believes its current approach of using ASME code is appropriate for this application. Additional justification for this conclusion will be included in the next revision of analysis and model report ANL–XCS–ME–000001, Design Analysis for the Ex-Container Components, to be completed prior to license application.

NRC Review: The DOE has proposed two different failure criteria for the waste package and drip shield, which are the principal components of the engineered barrier subsystem, when subjected to mechanical loading (e.g., static and dynamic rockfall loads, seismicity, igneous activity, and so on). The different engineered barrier subsystem component material failure criteria that have been proposed by DOE are (i) the 1992 ASME Boiler and Pressure Vessel Code (CRWMS M&O, 1996, Section 2.2.3) and (ii) the maximum normal stress theory (CRWMS M&O, 2000, Section 5.2.6.3). Neither of these two failure criteria consider the potential effects of base metal or welding flaws that may have been created during either the fabrication or emplacement of these engineered barrier subsystem components. Because of this deficiency, it may be more appropriate to use a fracture mechanics approach to assessing failures of the engineered barrier subsystem components subjected to mechanical loading.

Material failure, as defined in the ASME Boiler and Pressure Vessel Code, is based on a ductile failure process commonly referred to as plastic collapse. The onset of plastic collapse for metals can be approximated using several different stress threshold measures (i.e., Tresca stress, von Mises stress, Octahedral Shear stress, etc.). In the case of the ASME Boiler and Pressure Vessel Code, plastic collapse is defined in terms of the Tresca stress criterion, albeit indirectly. Whereas the Tresca stress, or maximum shear stress, is defined as the radius of the largest Mohr circle (Beer, et al., 1981, Section 6.6) for a given stress state at a point, the ASME Boiler and Pressure Vessel Code defines plastic collapse in terms of the diameter of the largest Mohr circle, which is referred to as the stress intensity (ASME International, 2001a, paragraph NB–3215). In terms of the principal stresses σ_1, σ_2, and σ_3; where $\sigma_1 \geq \sigma_2 \geq \sigma_3$, the stress intensity, *S*, can be represented mathematically as

$$S = \sigma_1 - \sigma_3 \qquad (1)$$

$$S_r = \frac{\sigma}{\sigma_c} \tag{2b}$$

From a fracture mechanics point of view, failure is assessed in terms of the material's fracture toughness, the applied stress, and the flaw size and geometry (Anderson, 1995). The fracture toughness of a given material is typically determined experimentally using applicable ASTM International standards and the plane strain fracture toughness should be viewed as a material property analogous to a material's yield and ultimate tensile strengths. Conversely, damage tolerant design and standard quality assurance engineering practices constrain the allowable applied stress that a structure may experience and the maximum flaw size and critical flaw geometry, density, and distribution within the structure. For example, the stress magnitude at the location of a flaw is dependent on the stress distribution within the structure in reaction to the applied load, which is controlled by the design of the structure, and any residual stresses created during the fabrication process. Similarly, the existing flaw sizes, geometries, densities, and distribution are largely determined and, in turn, controlled by the fabrication methods, including the concomitant quality assurance tests, used in constructing the structure.

DOE has opted to use a failure assessment diagram approach (Anderson, 1995, Section 9.4) to establish the failure criterion for assessing engineered barrier subsystem component material failure under mechanical loading.[1] Specifically, DOE has constructed failure assessment diagrams for compact tension, single-edge, notched-bend, and single-edge, notched-tension geometries and loading conditions. DOE asserts that these three geometries and loading conditions (i.e., plane stress and plane strain) should encompass all potential load states for the drip shield and waste package.

A failure assessment diagram is constructed using the following relationship (Anderson, 1995, Section 9.4)

where

K_I	—	stress intensity
K_{I_c}	—	fracture toughness (or critical stress intensity)
σ	—	applied stress (or applied load)
σ_c	—	plastic collapse stress (or failure load)

$$K_r = S_r \left[\frac{8}{\pi^2} \ln \sec \left(\frac{\pi}{2} S_r \right) \right]^{-1/2} \tag{2}$$

Referring to Figure 1, the structure will not fail from either fracture or plastic collapse if the locus of (S_r, K_r) is inside the safe failure assessment diagram failure envelope for a given applied

$$K_r = \frac{K_I}{K_{I_c}} \tag{2a}$$

load. If the locus of (S_r, K_r) lies outside of the failure envelope, the structure will fail under the applied load. If $S_r > 1$ and $K_r < 0.6$, then structural failure is dominated by plastic collapse and the ASME Boiler and Pressure Vessel Code failure criterion is applicable. Conversely, if $S_r < 0.4$ and $K_r > 1$, then structural failure is dominated by brittle fracture. For mixed mode

[1]Ziegler, J.D. "Transmittal of Report Addressing Key Technical Issue (KTI) Item Container Life and Source Term (CLST) 2.03." Letter (September 27) to J.R. Schlueter, NRC. Las Vegas, Nevada: DOE. 2002.

failure (i.e., when the characteristics of both plastic collapse and brittle fracture occur), failure cannot be predicted by either plastic collapse or brittle fracture alone. The use of a failure assessment diagram is significant in that it can be used to identify engineering applications where a mixed mode failure criterion needs to be considered.

To construct the failure assessment diagram failure curves for Titanium Grade 7 and Alloy 22, values for K_{I_c} and σ_c needed to be determined. K_{I_c} was approximated by the DOE using Charpy V-notch impact toughness data and the following correlating equation (ASME International, 2001b, Subarticle D–600)

$$\left(\frac{K_{I_c}}{S_y}\right)^2 = 5.0\left(\frac{CVN}{S_y} - 0.05\right) \qquad (3)$$

where

K_{I_c} — fracture toughness, ksi $\sqrt{\text{in}}$
S_y — yield strength, ksi
CVN — Charpy V-notch impact toughness, ft-lb

Anderson (1995, Example 9.1) recommended the effect of work hardening on the plastic collapse stress can be accounted for by averaging the yield and ultimate tensile strengths of the material. Specifically,

$$\sigma_c \cong \frac{S_y + S_u}{2} \qquad (4)$$

where

S_u — ultimate tensile strength

If it can be shown that $K_r \leq 0.6$ while $S_r \approx 1$ (Figure 1), or $K_r/S_r \leq 0.6$, then it can be reasonably assumed that failure of the structure will be attributable to plastic collapse and not brittle fracture. For the values of K_{I_c} and σ_c used for Titanium Grade 7 and Alloy 22, DOE demonstrated that $K_r/S_r \leq 0.6$ for all of the geometry and loading conditions considered (i.e., compact tension, single-edge, notched-bend, and single-edge, notched-tension). Note that a $K_r/S_r = 0.6$ ratio corresponds to a line rotated 31 degrees about the origin of the failure assessment diagram in a counter-clockwise direction from the abscissa.[2]

Although the basic approach to establishing the applicability of the plastic collapse and brittle fracture mechanics failure criteria is acceptable to the NRC staff, the estimates for the fracture toughness and plastic collapse stress for Titanium Grade 7 and Alloy 22 have yet to be adequately justified. Specifically, the justification for using Eq. (3) to estimate the fracture toughness of Titanium Grade 7 and Alloy 22 was not provided. As pointed out by Anderson (1995, Section 7.9.1), empirical correlations between Charpy V-notch impact toughness and fracture toughness "…seem to work reasonably well in some cases, but are unreliable in general." Moreover, Eq. (3) was developed specifically for correlating CVN and K_{I_c} for the upper shelf region of the brittle-ductile transition for pressure vessel steels (Barsom and Rolfe, 1970) and its applicability to titanium and nickel alloys has not been established. With regard to calculating the plastic collapse loads for Titanium Grade 7 and Alloy 22, DOE adjusted the ultimate tensile strengths of these materials from their engineering stress values to their Cauchy stress (i.e., true stress) counterparts. It is not clear that this conversion is justified within the context of a failure assessment diagram analysis [i.e., Eq. (2)] and additional documentation should be provided to substantiate this conversion.

It was suggested in the DOE response that Titanium Grade 24 would behave in a similar manner to Titanium Grade 7 and that no additional analyses would be necessary to assess the dominating failure mechanism for this material (i.e., plastic collapse or fracture). This assumption is not justified, however, given the significantly lower ductility Titanium Grade 24

[2]Ibid.

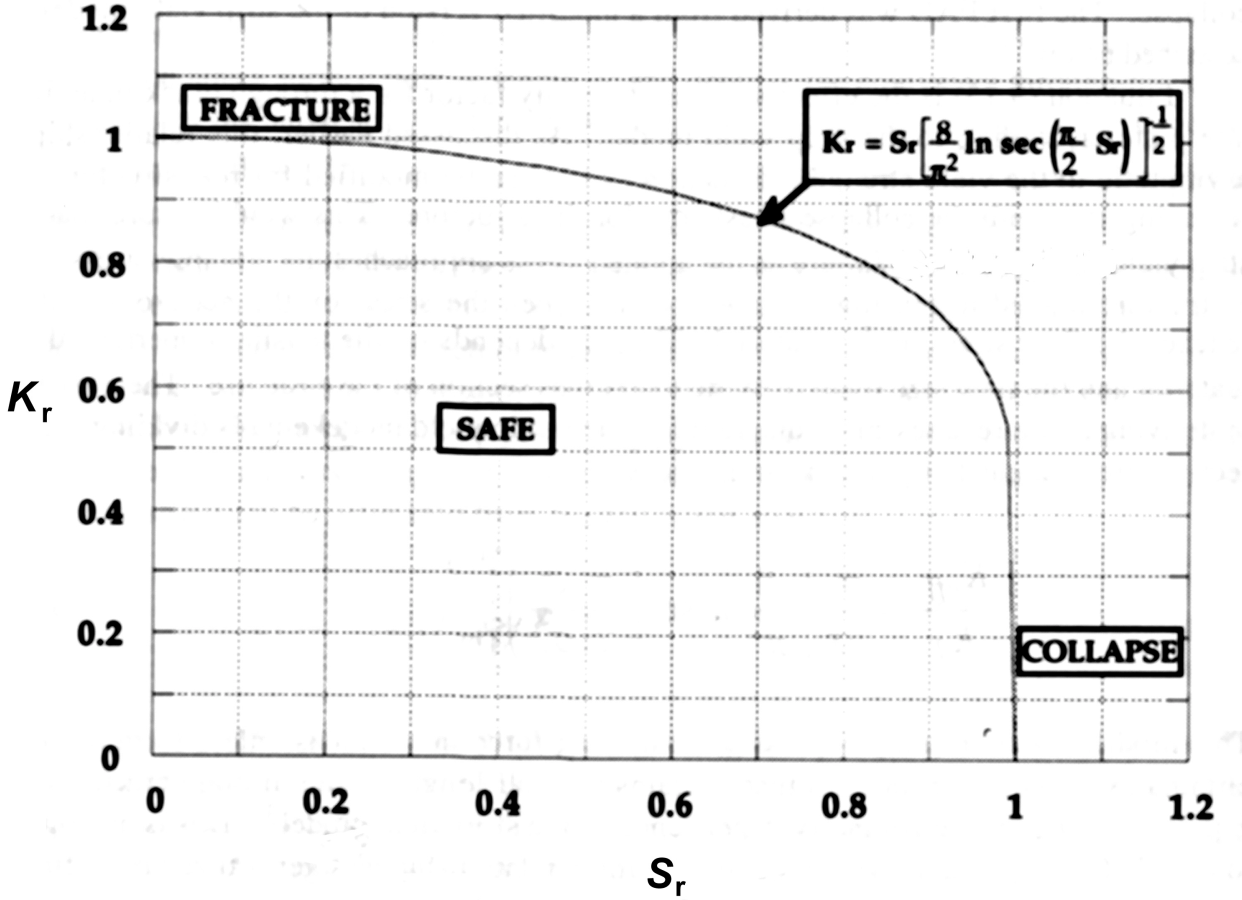

Figure 1. Failure Assessment Diagram (Anderson, 1995, Figure 9.12)

(i.e., 10 percent) as compared to Titanium Grade 7 (i.e., 20 percent) (ASTM International, 1998). Moreover, the yield strength for Titanium Grade 24 {828 MPa [120 ksi]} is much greater than that for Titanium Grade 7 {275 MPa [40 ksi]}, further indicating Titanium Grade 24 can be reasonably expected to behave in a more brittle manner than Titanium Grade 7. Furthermore, because of its dual phase (α and β) microstructure, the fracture toughness of Titanium Grade 24 will likely be affected differently than Titanium Grade 7 when subjected to the same thermal history (e.g., during the welding process).

DOE also asserted that the compact tension; single-edge, notched-bend; and single-edge, notched-tension geometries and loading conditions were representative of the drip shield and waste package conditions under mechanical loading. For the case of the drip shield subjected to rock block impacts, however, a combined mode of fracture may have to be considered for the Titanium Grade 7 plate near the Titanium Grade 24 bulkheads. This combined mode of fracture can be attributed to the high shear stresses that may occur in this region of the Titanium Grade 7 plate as the result of the rock block fracturing directly above the structurally stiff Titanium Grade 24 bulkheads. If it cannot be shown that the shear stresses in this region are small, the stress intensity, K_I, for this load scenario will have to be adjusted accordingly.

In addition to the foregoing, the Charpy V-notch impact toughness used to estimate the fracture toughness of Alloy 22 by way of Eq. (3) was derived from a test specimen that did not fracture completely (Haynes International, 1997). Although this observation might imply that the high ductility of Alloy 22 predisposes it to plastic collapse failure, there are fabrication processes and metallurgical mechanisms that could cause a more brittle deformation behavior (see Key

Technical Issue Agreements CLST.2.04, 2.05, 2.08, 6.02, 6.03, and PRE.7.03). These processes and mechanisms include (i) welding, which has been shown to reduce the Charpy V-notch impact toughness and yield strength of Alloy 22 (Edgecumbe Summers, et al., 2002); (ii) stress mitigation methods, such as laser peening and low plasticity burnishing; and (iii) the allowed variations in alloy composition that result in a loss of ductility or alter the precipitation kinetics of brittle secondary phases. A sensitivity study using a justifiable range of fracture toughness values that can account for these effects may be appropriate for establishing the governing failure mode of Alloy 22.

Additional Information Needs: The NRC has reviewed the DOE report addressing Key Technical Issue agreement item CLST 2.03. In general, the approach used by the DOE to establish the governing failure mechanism (i.e., brittle fracture or plastic collapse) for the drip shield and waste package materials using failure assessment diagrams is appropriate. The response to the agreement, however, failed to adequately establish the appropriate failure mechanisms for the drip shield and waste package materials because the various material properties used in the analyses have yet to be satisfactorily justified. To accomplish this task, the following additional information is needed by the NRC staff.

- Clarification of the material failure criterion expected to be used for assessing the response of the drip shield and waste package to mechanical loading (i.e., ASME Boiler and Pressure Vessel Code stress intensity or maximum normal stress theory). If the maximum normal stress theory criterion is to be used, its applicability to ductile metals must be justified.

- Justification of the fracture toughness values obtained from empirical correlations with Charpy V-notch impact toughness data.

- Justification for adjusting the ultimate tensile strengths from engineering stress to Cauchy stress (i.e., true stress) values.

- The effect of variations in the fracture toughness and plastic collapse stress of Titanium Grade 7 and Titanium Grade 24 on drip shield failure.

- Justification for not considering a combined mode of fracture failure in the Titanium Grade 7 drip shield plate when subjected to rock block impacts.

- The effect of fabrication and stress mitigation processes and allowed variations in alloy composition on the fracture toughness of Alloy 22.

Status of the Agreement: The agreement is categorized as “Needs Additional Information.”

REFERENCES

Anderson, T.L. *Fracture Mechanics, Fundamentals and Applications*. 2nd Edition. Boca Raton, Florida: CRC Press. 1995.

ASME International. *ASME Boiler and Pressure Vessel Code. Section III, Division 1—Subsection NB.* New York City, New York: ASME International. 2001a.

———. *ASME Boiler and Pressure Vessel Code*. *Section VIII, Division 3—Appendix D*. New York City, New York: ASME International. 2001b.

ASTM International. *Standard Specification for Titanium and Titanium Alloy Strip, Sheet, and Plate*. ASTM B 265-98. Philadelphia , Pennsylvania: ASTM International. 1998.

Barsom, J.M. and S.T. Rolfe. "Correlations Between K_{Ic} and Charpy V-Notch Test Results in the Transition Temperature Range." *ASTM STP–466: Impact Testing of Materials*. Philadelphia, Pennsylvania: ASTM International. pp. 281–302. 1970.

Beer, F.P. and E.R. Johnston, Jr. *Mechanics of Materials*. New York City, New York: McGraw-Hill, Inc. 1981.

CRWMS M&O. " Waste Package Design Methodology Report." ANL–EBS–MD–000053. Rev. 00. Las Vegas, Nevada: CRWMS M&O. 2000.

———. "Emplaced Waste Package Structural Capability Through Time Report." BBAA00000–01717–5705–00001. Rev. 00. Las Vegas, Nevada: CRWMS M&O. 1996.

Edgecumbe Summers, T.S., R.B. Rebak, T.A. Palmer, and P. Crook. "Influence of Thermal Aging on the Mechanical Properties and Corrosion Resistance of GTAW Welds of Alloy N06022." Scientific Basis for Nuclear Waste Management XXV. B.P. McGrail and G.A. Cragnolino, eds. Symposium Proceedings 713. Warrendale, Pennsylvania: Materials Research Society. pp. 45–52. 2002.

Haynes International. *Hastelloy Alloy C–22*. Kokomo, Indiana: Haynes International. 1997.